NRC PUBLIC MEETING SUMMARY REPORT

Date: January 12, 2012

Meeting Contact: Gary L. Stevens
RES/DE/CIB
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Gary.Stevens@nrc.gov

Subject: CATEGORY 2 PUBLIC MEETING – NUCLEAR POWER PLANT PIPING FATIGUE ISSUES

Meeting Date/Time: Thursday, January 5, 2012 / 9:00 AM

Location: U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Church Street Building, Room 06-B01
21 Church Street
Rockville, MD 20850

Purpose: The purpose of this meeting was to have technical discussions related to four topics in the area of reactor piping fatigue. The topics include:

1) Stress-based fatigue monitoring methodology for fatigue monitoring of Class 1 Nuclear Components in a Reactor Water Environment. EPRI has developed a proposed methodology to address the concerns of Regulatory Issue Summary (RIS) 2008-30. EPRI will provide an overview of the methodology and seek comments from the NRC and members of the public.
2) Improved Basis and Requirements for Break Location Postulation. EPRI has developed a tool that provides alternative bases and recommendations to be used in the event a break exclusion location reaches a cumulative usage factor (CUF) in excess of 0.1 during its operating lifetime. EPRI will explain the bases of this tool and obtain feedback from the NRC and members of the public on the approach used by the tool.
3) Process and Technical Basis for Identifying Environmentally Assisted Fatigue (EAF) Limiting Locations. EPRI has documented an approach that may be used to meet requirements for EAF evaluations using guidance from NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report." Some utilities are planning to use this approach, and may reference the document in future License Renewal Application submittals. EPRI will explain the bases of this approach and solicit feedback from the NRC and members of the public.
4) MRP-146, Revision 1, "Thermal Fatigue in Normally Stagnant Non-Isolable RCS Branch Lines." MRP-146 has been recently revised.

Since it is specifically referenced in the GALL Report, and consistent with past practice, EPRI would like to brief the NRC and members of the public of the changes made to this document.

Summary: The announcement and draft agenda for this meeting were posted on the NRC web site on December 6, 2011. They are available via ADAMS at Accession No. ML11336A152.

The final meeting agenda is included in **Attachment 1**.

Meeting attendance is included in **Attachment 2**.

Material presented at this meeting and referred to in the discussion below is included in the attachments to this meeting summary. The presentation material was originally submitted to the NRC via e-mail on January 3, 2012, which was subsequently posted in ADAMS at Accession No. ML12004A020; however, most of the presentations were subsequently updated and final copies were provided at the meeting. Therefore, the attached presentations supersede those contained in ADAMS at Accession No. ML12004A020.

Gary Stevens (NRC) opened the meeting at 9:00 AM with introductions, followed by the purpose of the meeting. It was noted that the NRC Office of Nuclear Regulatory Research (RES) currently has no on-going research activities in any of the areas being presented. It was noted that, although NRC RES is performing additional research on environmentally assisted fatigue (EAF), those activities do not include any efforts discussed in the presentations for this meeting.

Mo Dingler (Wolf Creek Nuclear Operating Company) provided a presentation (see **Attachment 3**) introducing the objective of the industry presentations that followed. In this presentation, the following was noted:

- EPRI/Industry/Materials Reliability Program (MRP) is presenting four technical topics related to fatigue of nuclear power plant components at this meeting to: (1) inform the NRC and public about these topics, and (2) solicit stakeholder input and comments on the approaches proposed to address these topics.

Tim Gilman (Structural Integrity Associates) provided a presentation (see **Attachment 4**) on Stress-Based Fatigue Monitoring: Methodology for Fatigue Monitoring of Class 1 Nuclear Components in a Reactor Water Environment. In this presentation, the following was noted:

- The subject EPRI report noted on the cover of the presentation is publicly available via the EPRI website.
- The industry presentation is summarized as follows:
    - EPRI-sponsored methodology was presented for stress-based environmental fatigue monitoring which addresses RIS 2008-30.
    - Overall methodology combines many proven practices.
    - Basic steps in the process include:
        - Multiaxial stress calculations
        - Address NRC RIS 2008-30 and RIS 2011-14
        - Accurate knowledge of through-wall distributions
    - Smart Peak/Valley Detection and Stress Cycle Counting
        - Rubberband (detects reversal regions using multiaxial stress range criteria)
        - Rainflow-3D (identifies stress cycles)
    - Calculation of EAF
        - Meets GALL Report requirements
        - Implements Expert Panel guidance

Terry Herrmann (Structural Integrity Associates) provided a presentation (see **Attachment 5**) on Improved Basis and Requirements for Break Location Postulation. In this presentation, the following was noted:

- The subject EPRI report noted on the cover of the presentation is publicly available via the EPRI website.
- The industry presentation is summarized as follows:
    - The current cumulative usage factor (CUF) criterion of 0.1 for postulated break locations has no clear technical basis.
    - Continued use of this criterion could result in unnecessary costs without an associated safety benefit.
    - Over four decades of industry experience have demonstrated that design transients do not result in high energy line breaks.
    - Industry experience has been used to address uncertainties that existed when the current CUF criterion was established.
    - Five components were selected from NUREG/CR-6674 for evaluation, which provided a range of loads, material types and reactor designs.
    - Use of leak probabilities (versus rupture) is conservative when considering postulated high energy line breaks.
    - Initiation and leak probability calculations based on NUREG/CR-6909 were performed using pc-PRAISE.

    - Core Damage Frequency is related to the leak frequency, consistent with the methodology used in NUREG/CR-6674.
    - Resulting Core Damage Frequency (CDF) vs. EAF CUF (CUF_{en}) plots show no direct correlation between CUF_{en} and CDF values.
    - Many current plants are designed to ANSI/ASME B31.1, which does not require calculation of CUF.
    - For all of the selected components, a CUF_{en} of 1.0 resulted in a CDF $\leq 1x10^{-6}$, which NRC Regulatory Guide 1.174 considers very small and is well below the $1x10^{-4}$ value promulgated in NUREG-0800 Chapter 19.
    - Based on the results of this study, if the use of CUF as a break location criterion is to be continued in combination with environmental fatigue analysis, a CUF_{en} of 1.0 can be used without an impact to safety.
    - An approach that applies both deterministic and probabilistic elements is proposed.
    - The proposed methodology is consistent with the NRC policy statement for use of PRA methods and the principles outlined in Regulatory Guide 1.174.
    - Branch Technical Position (BTP) 3-4 should be revised to apply this methodology.
- The NRC noted the following:
    - NRC staff explained to the meeting participants the regulatory rationale behind these guidelines. Refer to **Attachment 8**.

Dave Gerber (Structural Integrity Associates) provided a presentation (see **Attachment 6**) on Environmentally-Assisted Fatigue Screening. In this presentation, the following was noted:

- The subject EPRI report noted on the cover of the presentation will be publicly available via the EPRI website. It is currently in draft form and is scheduled for completion in February 2012.
- The proposed methodology is intended to address issues raised in GALL Report Rev. 2.
- The industry presentation is summarized as follows:
    - A technical basis of the screening process used to evaluate a plant to determine EAF limiting locations for fatigue monitoring was provided.
    - The screening process is designed to equip license renewal applicants with a consistent method to identify EAF limiting locations additional to the sample locations

evaluated in NUREG/CR-6260 for their reactor type and vintage.

- Guiding principles for the screening and ranking process include:
  - Consistent technical basis.
  - Analytical method using readily available design input from process and instrumentation diagrams (P&IDs), piping isometric drawings and piping stress reports.
  - Only basic stress or fatigue analysis required.
- The following are the basic areas of new technology developed by this project:
  - Procedure for Estimating F_{en} Factors.
  - Procedure for Estimating U_{en}.
- An example of the process will be provided in the supporting technical report.

Bob McGill (Structural Integrity Associates) provided a presentation (see **Attachment 7**) on Thermal Fatigue Management Guideline for Normally Stagnant Non-Isolable RCS Branch Lines. In this presentation, the following was noted:

- The subject EPRI report noted on the cover of the presentation is not publicly available.
- The industry presentation is summarized as follows:
  - MRP-146 Revision 1 allows for progressively more specific and rigorous evaluation as part of the assessment process
    - General screening
    - Determine significance of thermal fatigue potential
    - Inspection frequency based on severity of loading
  - Many conservatisms inherent with each level
  - MRP-146 Revision 1 provides utilities with the most current implementation guidance (replacing Revision 0).
  - MRP-146 Revision 1 and supporting documents provide an effective approach to managing thermal fatigue in normally stagnant, non-isolable reactor coolant system (RCS) branch lines
  - Pressurized water reactor (PWR) owners are using this approach moving forward
  - EPRI committed to keeping the guidance current through future revision based on owner operating experience

There were no other presentations offered, nor were there any comments from any members of the public.

Finally, there was brief discussion of all of the meeting presentations. NRC asked if there was any activity planned for the simplified EAF analysis approach presented by the industry in a Public Meeting held on March 22, 2011. The industry responded that there were no current plans to pursue that activity further at this time.

No actions were identified for this meeting. The NRC announced that there are no currently planned future public meetings on this topic. Interested stakeholders were thanked for their briefings, and were encouraged to continue future dialogue and to let the NRC know of any additional need to meet.

The meeting was adjourned at approximately 3:00 PM.

Attachments: The following attachments are included with this report:

Attachment 1

AGENDA

NUCLEAR POWER PLANT PIPING FATIGUE ISSUES

January 5, 2012

9:00 AM – 3:00 PM EST

Location:

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Church Street Building, Room 06-B01
21 Church Street
Rockville, MD 20850

Purpose of Meeting:
The purpose of this meeting is to obtain feedback from interested technical parties on thermal fatigue-related topics related to operating nuclear power plant piping.

Time	Topic	Organization	Coordinator or Presenter
9:00	Welcome and Introductions	NRC	Stevens
9:10	Purpose of Meeting	NRC	Stevens
9:15	EPRI NRC Nuclear Power Plant Piping Fatigue Issues Meeting Introduction	MRP/WCNOC	Dingler
9:20	Stress-Based Fatigue Monitoring: Methodology for Fatigue Monitoring of Class 1 Nuclear Components in a Reactor Water Environment	SIA	Gilman
10:15	**BREAK**		
10:30	Improved Basis and Requirements for Break Location Postulation	SIA	Herrmann
11:30	Environmentally-Assisted Fatigue Screening	SIA	Gerber
12:30	**LUNCH**		
1:30	Thermal Fatigue Management Guideline for Normally Stagnant Non-Isolable RCS Branch Lines	EPRI SIA	Chu McGill
2:30	Public Comments	NRC	Stevens
2:45	Summary and Actions	NRC	Stevens
3:00	**ADJOURN**		

Attachment 2
ATTENDANCE LISTS

The individuals listed on the following 3 pages attended the meeting.

The following individuals announced their participation via teleconference:

Name	Organization	E-mail
Al Butcavage	NRC	alexander.butcavage@nrc.gov
On Yee	NRC	on.yee@nrc.gov
Rich Schaller	Enterprise Technical Services	RFSchaller@enterprisetec.com
Glenn Michael	Arizona Public Service	glenn.michael@aps.com
Terry Childress	Duke Energy	terry.childress@duke-energy.com

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(Table)

ATTENDANCE LIST
for
Public Meeting

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9:00 AM -- 3:00 PM EST

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Tim Gilman	"	tgilman@structint.com
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Shannon Chu	EPRI	schu@epri.com
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MIKE McDEVITT	SoCalEdison	mike.mcdevitt@sce.com
Bob McGill	Structural Integrity Assoc.	rmcgill@structint.com
Gary Stevens	NRC/RES	gary.stevens@nrc.gov

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Name	Organization	E-mail
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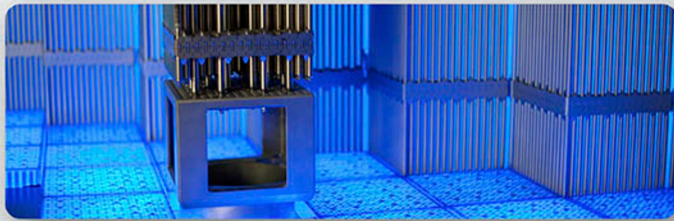
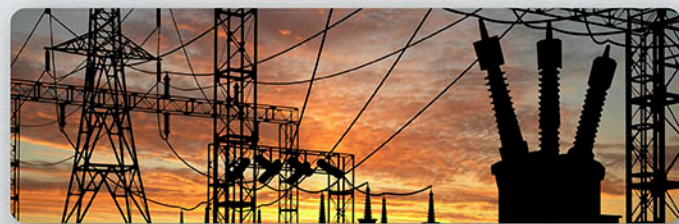
Attachment 3

EPRI NRC NUCLEAR POWER PLANT PIPING FATIGUE ISSUES MEETING
INTRODUCTION PRESENTATION


EPRI NRC Nuclear Power Plant Piping Fatigue Issues Meeting Introduction

Mo Dingler

MRP TS TAC Chairmen

Public Meeting

January 5, 2012

Topics and Objectives (1/2)

- Stress- Based Fatigue Monitoring Methodology to address RIS 2008-030
  - Discusses methodology for including 6-stress tensors and including EAF
  - Identify an approach addresses the concern identified in the RIS

- Improved Basis and Requirements for Break Location Postulation
  - Share results of work that shows CUF does not correlate well with leak probability and consequence of CUF_{en}=1.0 is low, alternate approach offered
  - Identify alternatives to the current CUF approach that may be applied in a situation where a break exclusion location reaches a CUF above 0.1

Topics and Objectives (2/2)

- Process and Technical Basis for Identifying Environmentally Assisted Fatigue (EAF) Limiting Locations
  - provides an approach for identifying most limiting locations as specified for license renewal under GALL 2
  - Identify an approach that meets the intent of the GALL 2 requirements
- Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines (MRP-146, Revision 1)
  - Update NRC on changes since MRP-146S was presented
  - Potential NRC action : GALL 2 references MRP-146 Rev. 0 which is now a superseded document, recommend interim staff guidance to address this issue

NEI 03-08 Materials Initiative (Industry Implementation of MRP-146)

- *MRP-146 (and S and Rev. 1) were issued under the Materials Initiative with "Needed" Requirements*
- *The purpose of this Initiative is to:*
  - *provide a consistent management process*
  - *provide for prioritization of materials issues*
  - *provide for proactive approaches*
  - *provide for integrated and coordinated approaches to materials issues*
- *Actions required by this Initiative include:*
  - *commitment of executive leadership and technical personnel*
  - *commitment of funds for materials issues within the scope of this Initiative*
  - *commitment to implement applicable guidance documents*
  - *provide for oversight of implementation*

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Environmentally Assisted Fatigue Expert Panel (Industry Input to Additional Fatigue Topics)

- Objective of panel is to provide leadership for industry activities to address EAF, identify research needs and set priorities
- Objective of research is to minimize the impact of any new procedures and acceptance criteria for the plant owners while meeting the NRC regulation goals for extended plant life and new plants
- EAF Panel first met in June 2010
- MRP funded EPRI research on this issue since ~2005
- BWRVIP and ANT also started funding EPRI research on EAF issues in 2011
- Panel participants include NSSS and A/E vendor organizations, Utility Staff, and ASME members (independent contractors)
- NRC (Regulation and Research) representatives have attended and contributed
- Regular meetings held during ASME BPVC meetings

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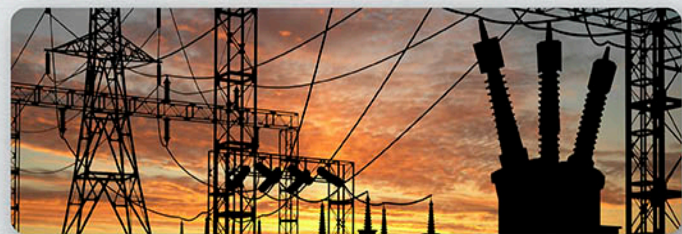
Attachment 4

STRESS-BASED FATIGUE MONITORING: METHODOLOGY FOR FATIGUE MONITORING OF CLASS 1 NUCLEAR COMPONENTS IN A REACTOR WATER ENVIRONMENT PRESENTATION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Stress-Based Fatigue Monitoring: Methodology for Fatigue Monitoring of Class 1 Nuclear Components in a Reactor Water Environment

(EPRI technical report 1022876)

Tim Gilman
Associate

NRC Fatigue Meeting
January 5, 2012

Presentation Objective

- Present EPRI-sponsored methodology for stress-based environmental fatigue monitoring which addresses RIS 2008-30.
- Obtain concurrence from NRC that general approach outlined here resolves the concerns expressed in the RIS.

General Objectives of EPRI Report

- Resolve regulatory concerns about use of single stress term in fatigue monitoring (RIS 2008-30).
- Provide for automatic calculation of environmentally-assisted fatigue (EAF) – not just ASME Code fatigue.

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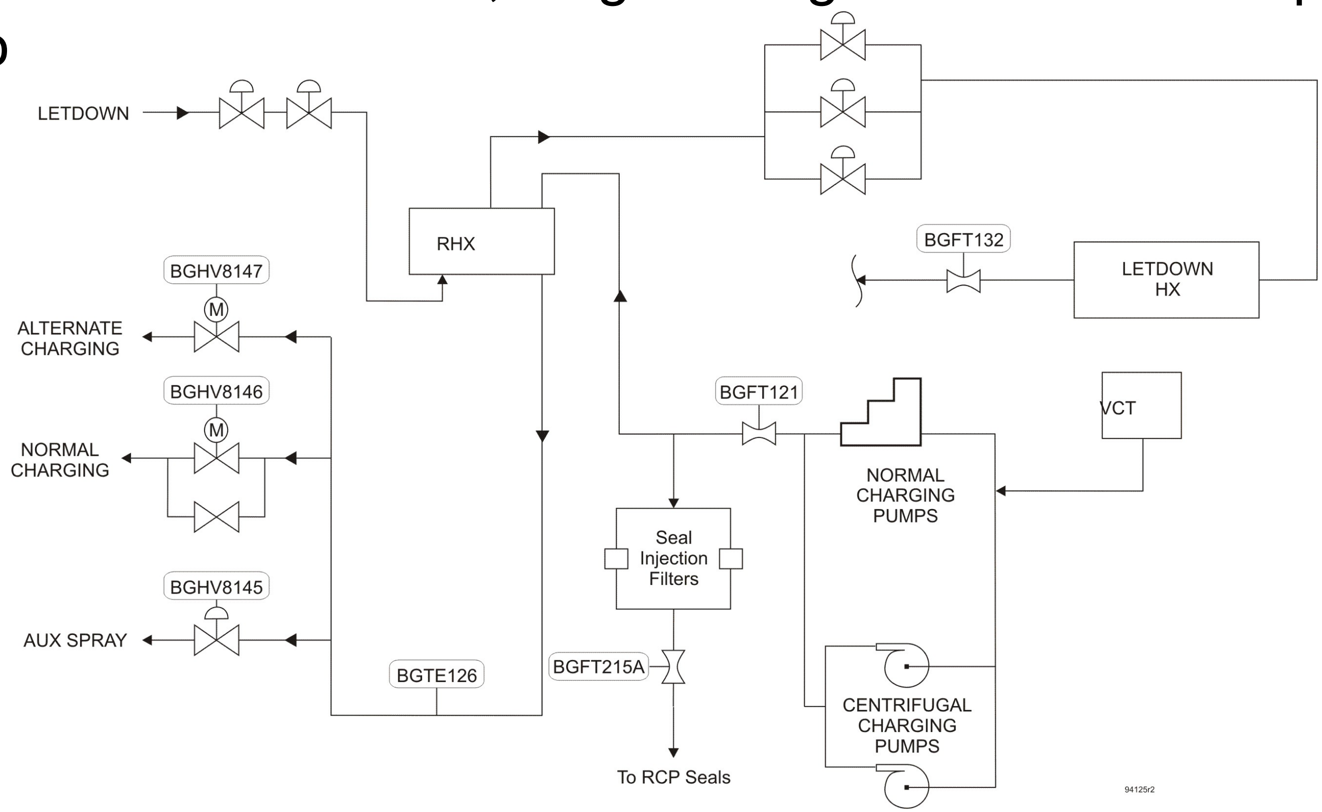
NRC Position on Fatigue

- Draft NRC RIS-2008-XX ("Fatigue Analysis of Nuclear Power Plant Components" May 2008) was issued to inform licensees of NRC staff concern about use of simplified single stress term in fatigue evaluations.
- NRC responded to public comments on the draft RIS in Dec 2008.
- Final RIS-2008-30 issued Dec 2008.
- Fatigue calculations must <u>consider</u> all six stress components in accordance with ASME Subarticle NB-3200 guidance.

What is Stress-Based Fatigue (SBF)?

- Actual plan measured data (temperatures, pressures, flow rates, valve positions, etc.) are used to compute detailed stress histories.
- From the stress histories, fatigue usage factors are computed for mo



FatiguePro and RIS-2008-30

- Historically, single stress term sometimes used for fatigue evaluations
  - Originally necessary because of computer limitations
  - Conventional stress cycle counting algorithms use single stress
  - Simplified methodology can be shown to be conservative, but great deal of judgment may be required for development
- Subsequent RAIs related to fatigue analysis question analyst judgments involved in general.

Guiding Principles for Development

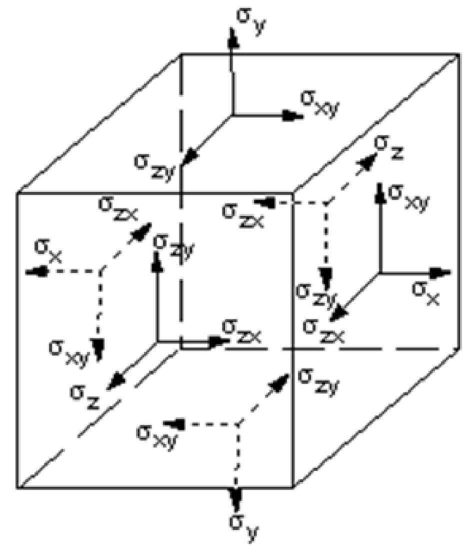
- Accuracy
  - Benchmarks reproduce known problems
  - Meet design basis (ASME Subarticle NB-3200) and regulatory requirements (NUREG-1801, GALL Report)
  - Industry guidance (EPRI's EAF Expert Panel lessons)
- Validation
  - Results make physical sense.
  - Consistent with sound science and engineering principles.
- Repeatability
  - Comply with ASME NQA-1
  - Minimize analyst and user judgments
- Transparency
  - Technical basis documented in EPRI report
  - Available for everyone to review

SBF Technical Basis
Significant Areas of New Technology

- Stress calculations (linearized membrane, bending, and peak components)
- Stress peak and valley detection
- Stress cycle pairing and fatigue calculations
- Environmental fatigue (F_{en}) calculations

Stress Calculation Objectives

- Compute time history of the 6 unique stress components for:
  - Primary plus secondary (usually linearized membrane plus bending)
  - Total stresses (including peak)
- Plus metal surface temperature



	Primary (P) + Secondary (Q)						Primary (P) + Secondary (Q) + Peak (F)						
Time	SX	SY	SZ	SXY	SYZ	SXZ	SX	SY	SZ	SXY	SYZ	SXZ	Temp
0	2.281	47.413	37.257	1.772	0.000	0.000	0.128	72.722	63.606	0.097	0.000	0.000	154.3
1	2.291	47.830	37.637	1.762	0.000	0.000	0.130	73.226	64.094	0.096	0.000	0.000	149.4
2	2.301	48.245	38.014	1.752	0.000	0.000	0.131	73.726	64.579	0.095	0.000	0.000	144.6
3	2.313	48.607	38.301	1.755	0.000	0.000	0.132	74.181	64.973	0.095	0.000	0.000	142.8
4	2.323	48.972	38.598	1.756	0.000	0.000	0.134	74.638	65.376	0.094	0.000	0.000	140.5
5	2.333	49.357	38.939	1.749	0.000	0.000	0.135	75.053	65.767	0.093	0.000	0.000	136.6

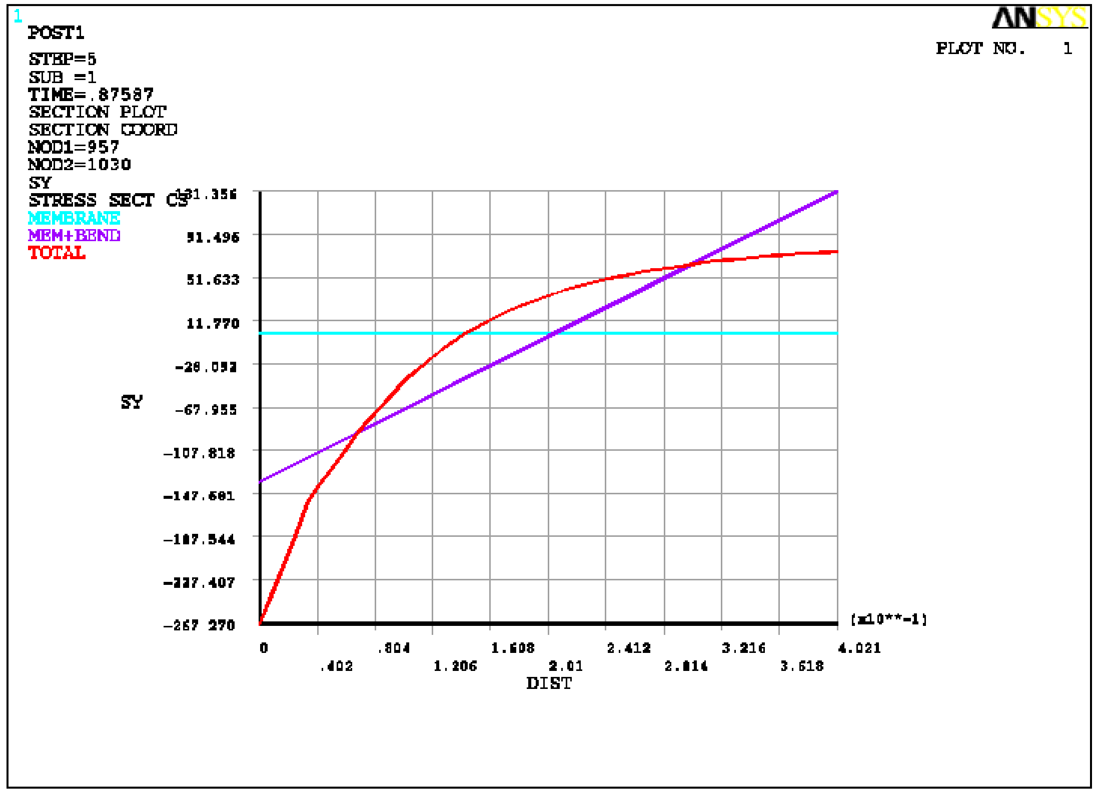
Stress Calculations (cont'd)

- Linearized stresses for static loads are scalable
  - Pressure
  - Piping interface loads (forces, moments)
- Thermal (time-dependent) stresses are calculated with Green's Functions
  - Green's Functions are simply influence functions
  - The RIS clearly states, "The Green's function methodology is not in question."

Linearized Thermal Stresses

- Requires use of appropriately conservative ratio of (P+Q) to (P+Q+F) (*analyst judgment or previously performed fatigue analysis*), OR

- Accurate knowledge of time-dependent, through-wall stress distribution



Implemented the latter to improve accuracy and minimize analyst judgments.

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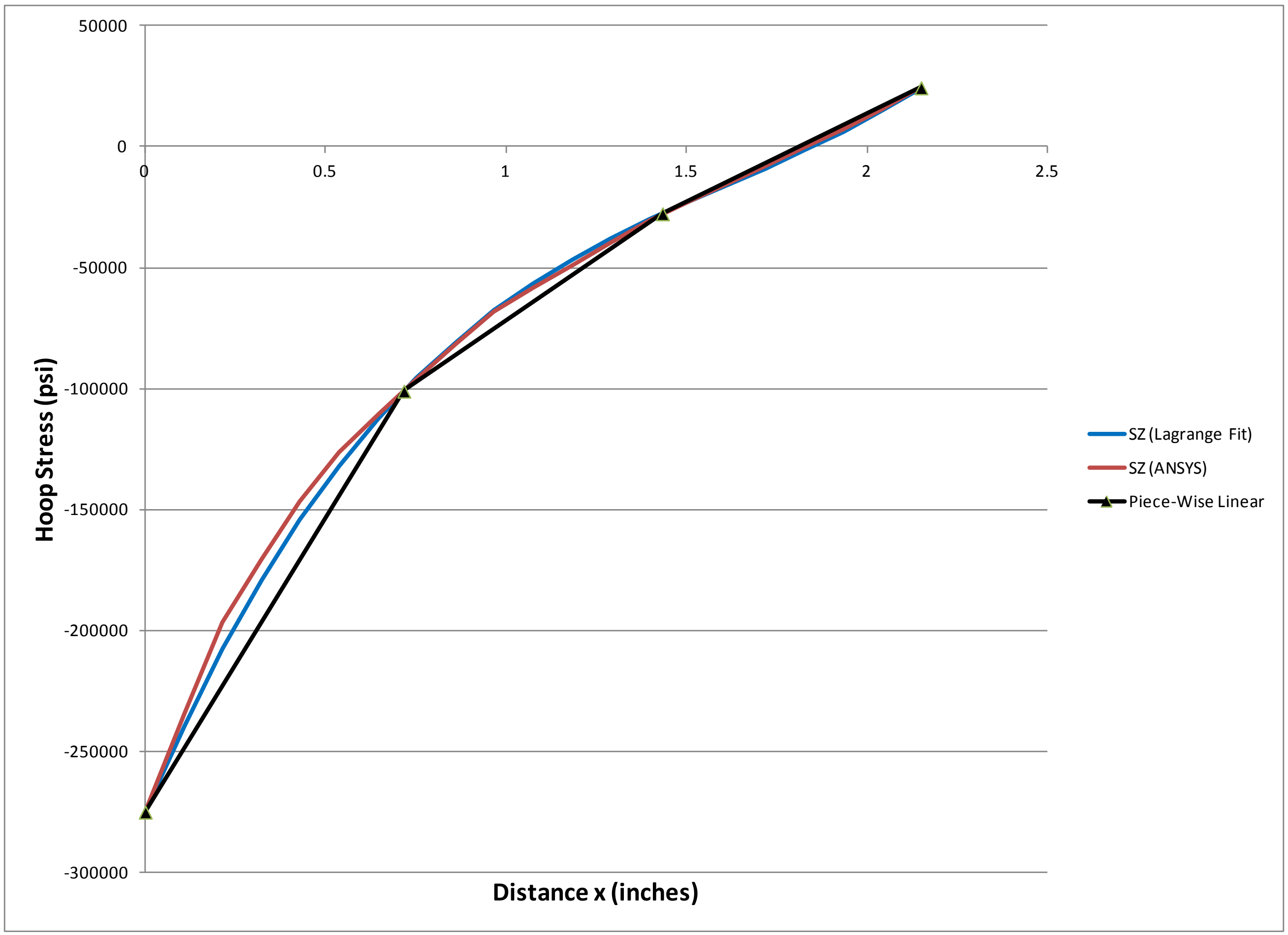
Linearized Thermal Stresses (cont'd)

- Use of either Lagrange Polynomial: $y = H_0 + H_1 x + H_2 x^2 + \ldots + H_p x^p$
- or piece-wise linear distribution



Linearized Thermal Stresses (cont'd)

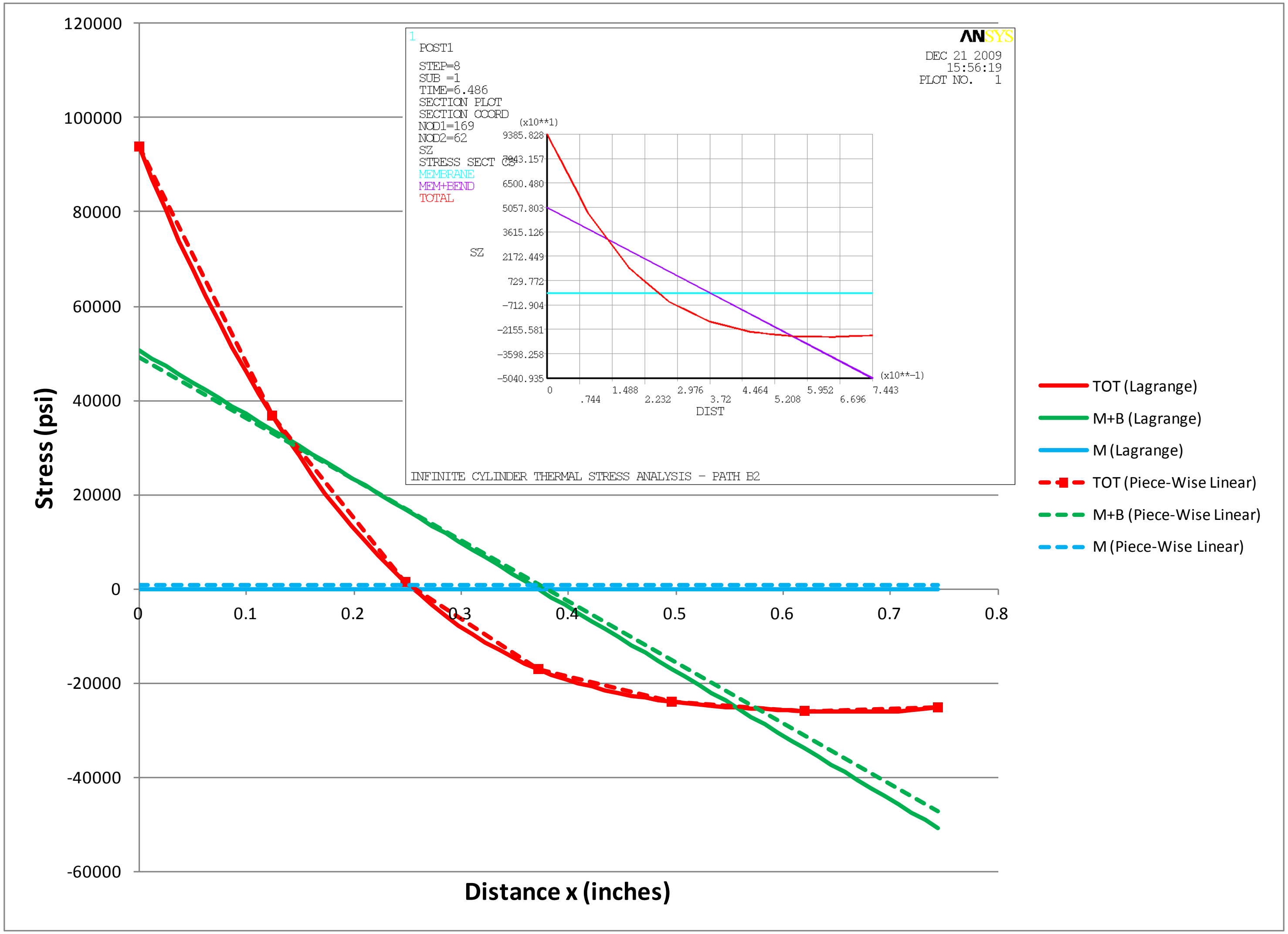
- Conventional membrane and bending stress computations.
  - "Cartesian" or generalized linearization in ANSYS
  - "Linearization for three-dimensional structures" in ABAQUS
  - Stress Linearization Procedure described in Section 5.A.4.1.2 of ANNEX 5.A of ASME Section VIII, Division 2

$$\sigma_m = \frac{1}{t}\int_0^t \sigma(x)\,dx$$

$$\sigma_b = \frac{6}{t^2}\int_0^t \sigma(x)\left(\frac{t}{2} - x\right)dx$$

Closed form solutions using previously determined Lagrange Polynomial

Linearized Thermal Stresses (cont'd)



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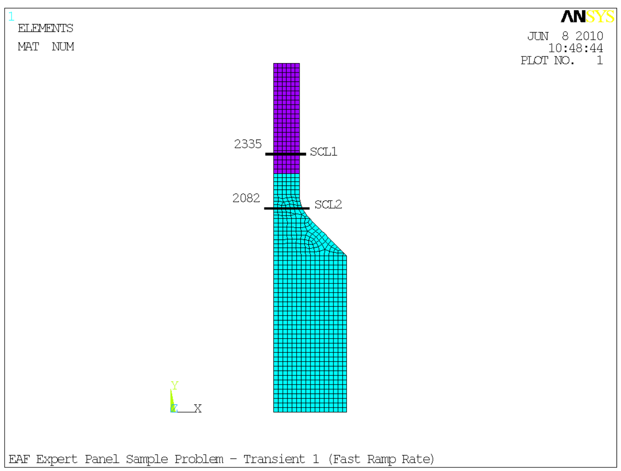
Multiaxial Green's Function

Macro developed to create multiaxial Green's Function.

Includes all information necessary to compute linearized thermal stresses at a stress classification line for a given film coefficient.

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! Created by tgilman from GRNSFN-S.rst   14:33:54
2335 ! Node 1
2327 ! Node 2
8 ! P = polynomial degree = # of thru-wall divisions
24 ! NumSet = number of time blocks in Green's Function
0 ! RSYS (results coordinate system)
1.406 ! Path Length
5.594 14 0 ! XG YG ZG (Global Coordinates Inside Node)
7 14 0 ! XG YG ZG (Global Coordinates Outside Node)
70.  653.   ! temperature range
100.   ! flow
1.E-05 ! Time
-1.817551361E-13 6.412294996E-11 1.330949231E-10 -2.133951005E-13 0 0 70
-1.182986039E-12 5.69802081E-11 1.181114205E-10 -6.42931852E-12 0 0 70
-7.523217209E-12 4.102893909E-11 8.667579707E-11 -1.237429329E-11 0 0 70
-1.092514775E-11 1.866163099E-11 5.190668012E-11 -1.527801468E-11 0 0 70
-3.111476776E-12 -7.729228141E-12 2.692269684E-11 -1.237017793E-11 0 0 70
5.560239026E-12 -1.77459226E-11 2.218901776E-11 -9.715285252E-12 0 0 70
5.860479508E-12 -3.308940149E-11 5.507489079E-12 -7.069777321E-12 0 0 70
2.858558287E-12 -4.647084963E-11 -9.696455806E-12 -3.392061972E-12 0 0 70
1.623788983E-12 -5.060145187E-11 -9.997383495E-12 2.359452966E-12 0 0 70
1.00001 ! Time
-445.518934 -84935.4003 -84453.4724 1.94566036 0 0 365.101373
-1137.78108 -12282.4275 -11023.828 1.63401116 0 0 87.3745052
-1140.6189 9149.21116 10485.4693 5.2931554 0 0 73.8726438
-835.950981 6254.25106 7352.8488 1.38770485 0 0 70.5824588
-605.695909 5927.42761 6856.73975 -21.6177287 0 0 70.0687223
-438.091397 5857.7175 6618.81317 -35.4436409 0 0 70.0040231
-287.269029 5790.13841 6400.45774 -33.2671749 0 0 70.0008471
-144.514695 5709.609 6195.58535 -19.6472641 0 0 70.000149
-1.11428874 5601.04792 5998.10791 0.857158159 0 0 70.0000296
1.99945 ! Time
-544.856995 -116254.633 -115319.552 0.924715645 0 0 443.847375
-1776.87995 -24480.513 -22233.6017 -32.6231612 0 0 136.819644
-1879.19238 8247.95241 10685.0167 -44.7392203 0 0 81.5690501
```


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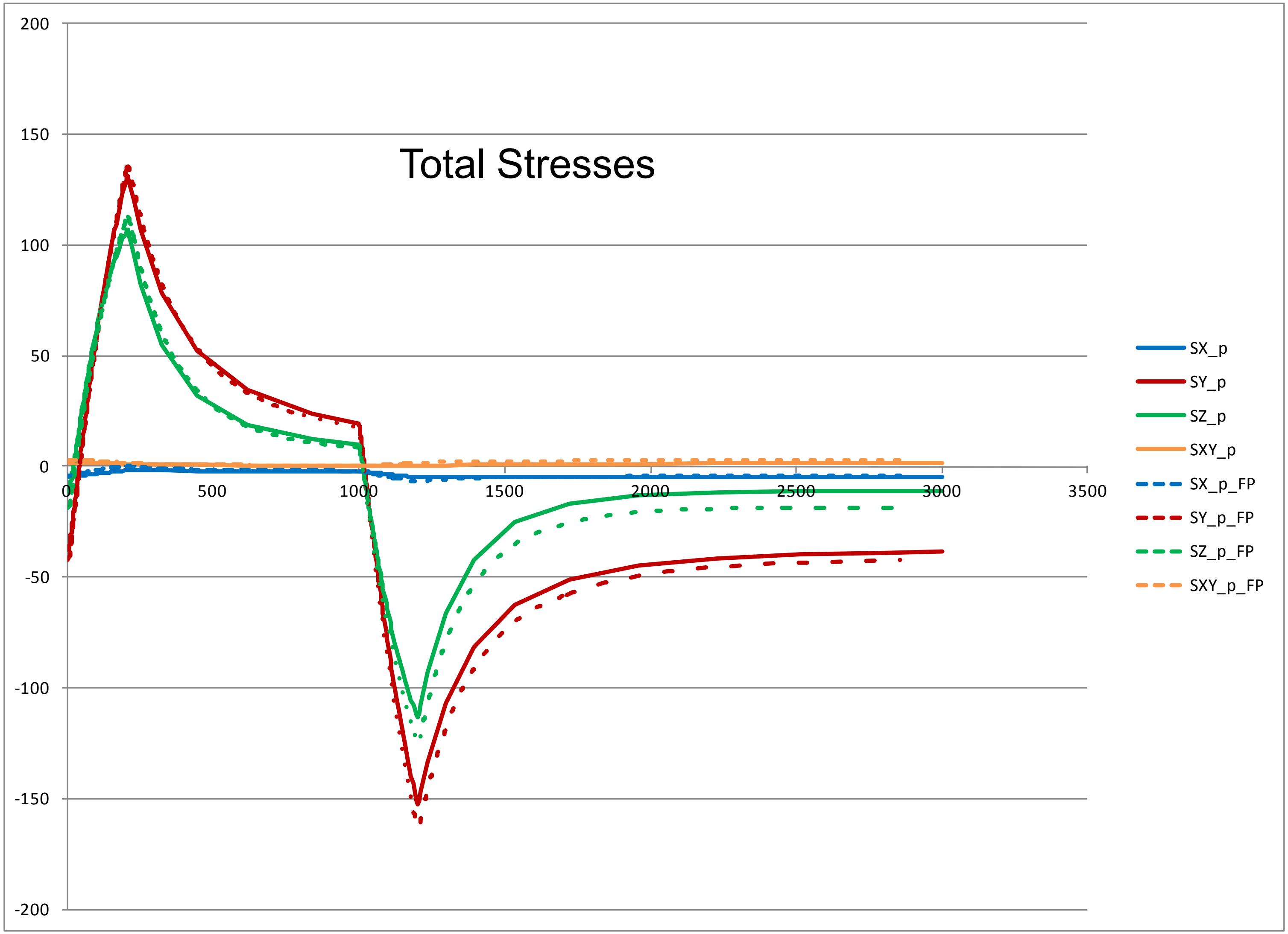
Computation of Primary Plus Secondary and Total Stresses

- Pressure, piping interface, and other "static" loads added together by scaling to pressure, temperature, etc.
  - M+B and Total
- Through-wall time-dependent thermal stresses computed using Green's Functions
  - All six M+B components computed based on through-wall distributions
- Fatigue strength reduction factors applied on M+B stresses, as appropriate
- Thermal peak is superimposed after the FSRF/SCF is applied.
- Same methodology implemented as in the EPRI EAF Expert Panel sample problem.

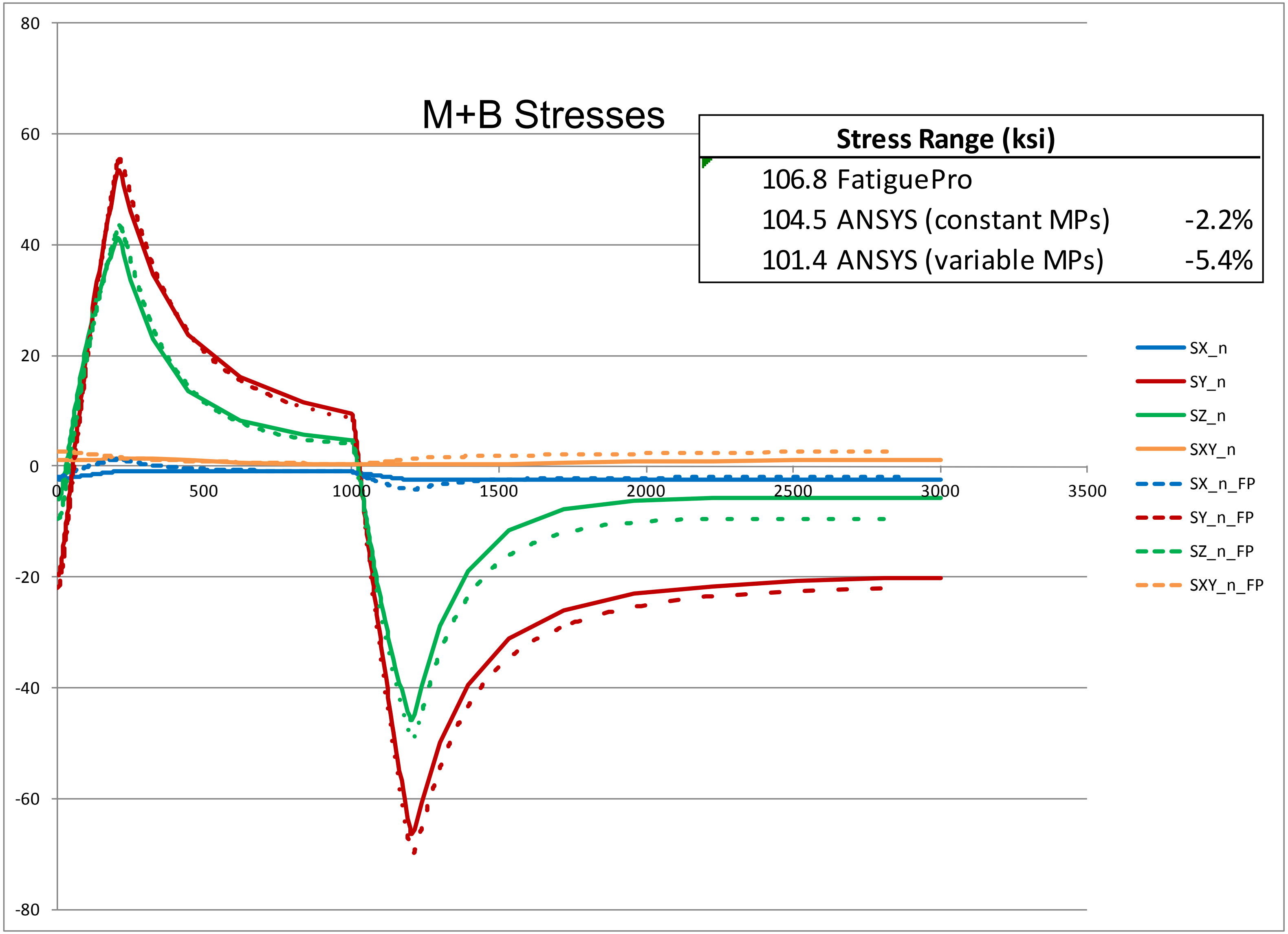
Benchmark of Stress Calculations

- Sample problem from EPRI’s EAF Expert Panel performed
- Comparisons made to values computed with ANSYS using temperature-dependent material properties.
  - Total stress
  - Membrane plus bending stress
  - Temperature

Benchmark of Total Stresses



Benchmark of Linearized Thermal Stresses



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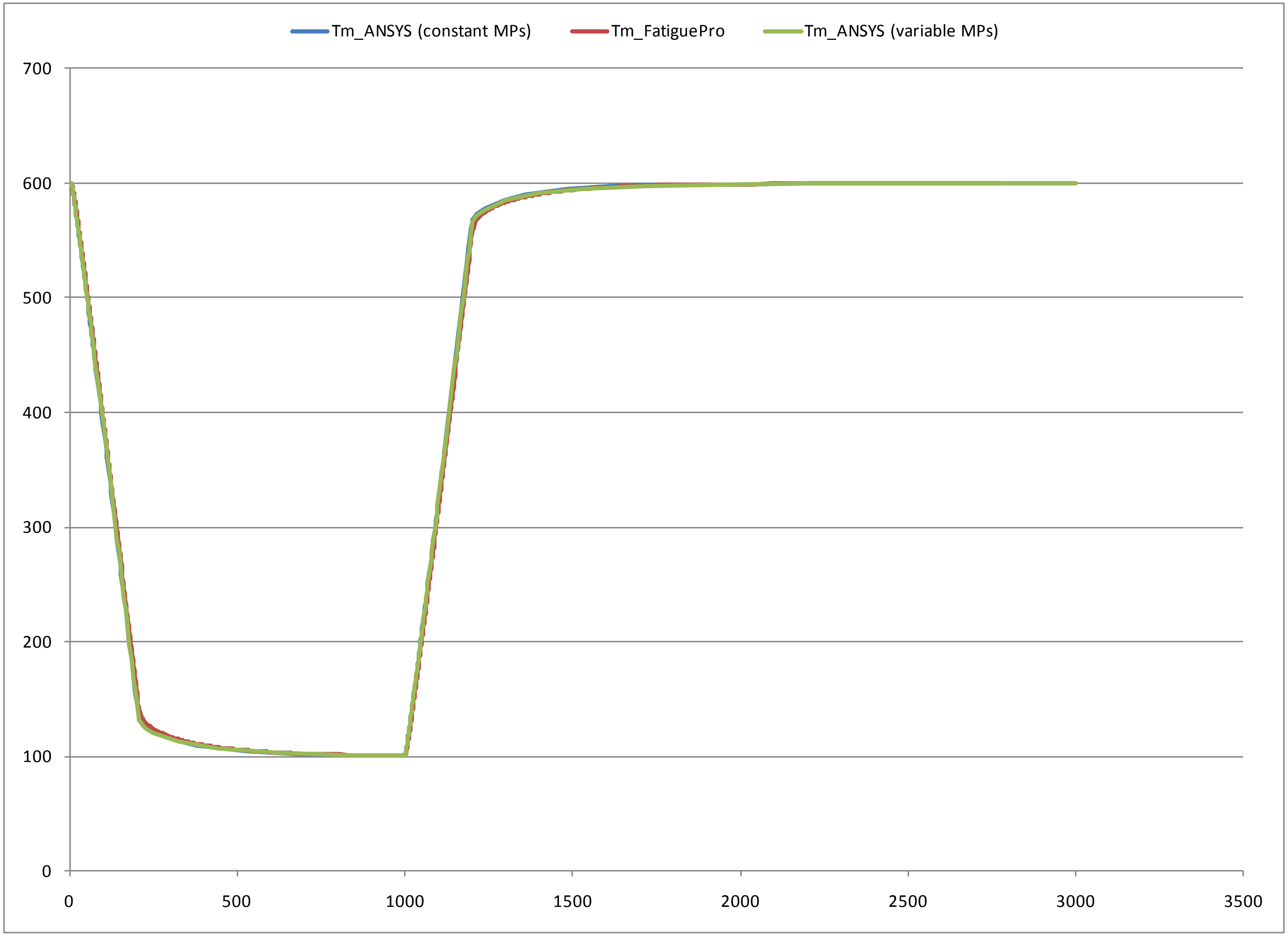
Benchmark of Metal Temperature Calculations



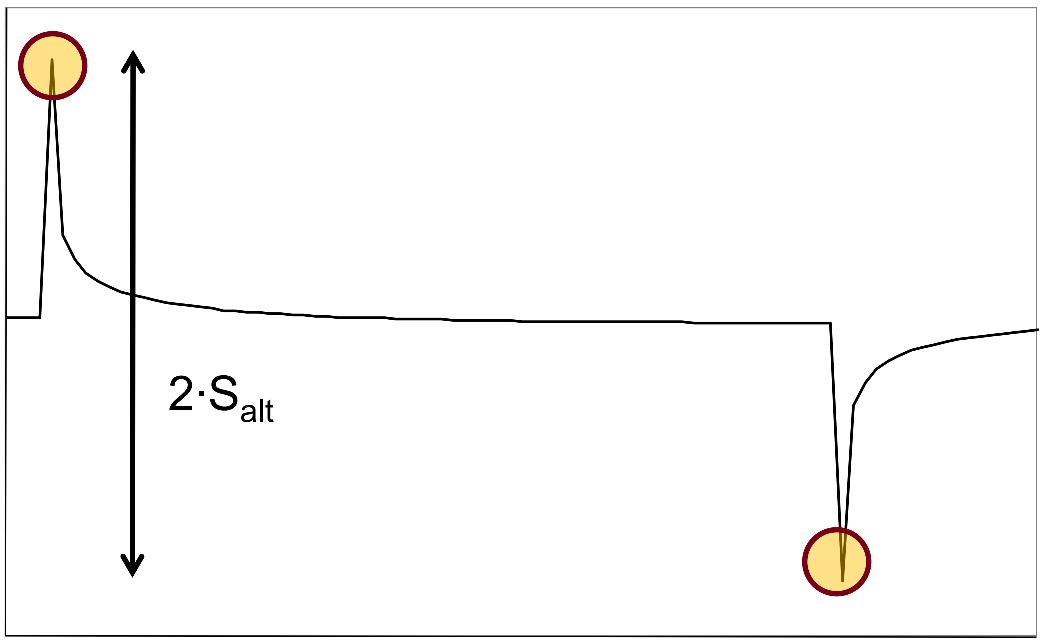
Stress Cycle Counting
Design vs. Monitoring

- Design Assumptions
  - Idealized transient definitions
  - Maximum number of cycles
  - Cycles postulated to occur in worst-possible order
- Monitoring
  - Real data
  - Typically less severe stress ranges, but increased complexity
  - Cycles are known to occur in actual order

Taking order into account using all six stress components and other ASME Code rules requires a non-trivial solution!

Idealized ASME "Stress Cycle"

"… a condition where the alternating stress difference [NB-3222.4(e)] goes from an initial value through an algebraic maximum value and an algebraic minimum value and then returns to the initial value."



An "operational cycle" can contain multiple stress cycles.

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Stress Cycles
Traditional Design Analysis Example

Transient order unknown
Local extreme stress conditions (peaks or valleys) assumed to pair in worst-possible order.

Stress Cycles Monitoring Example

- Local extreme conditions evaluated in known order.
- One large stress reversal with multiple internal cycles.
- Factor of 10 (possibly more, depending on analyst judgment) difference using known order.

Sp	Ke	Salt	n	Nallow	U
200	3.333	333.3	1	54.47	0.01835873
50	1	25	18	1481072	1.21534E-05
				CUF =	0.018

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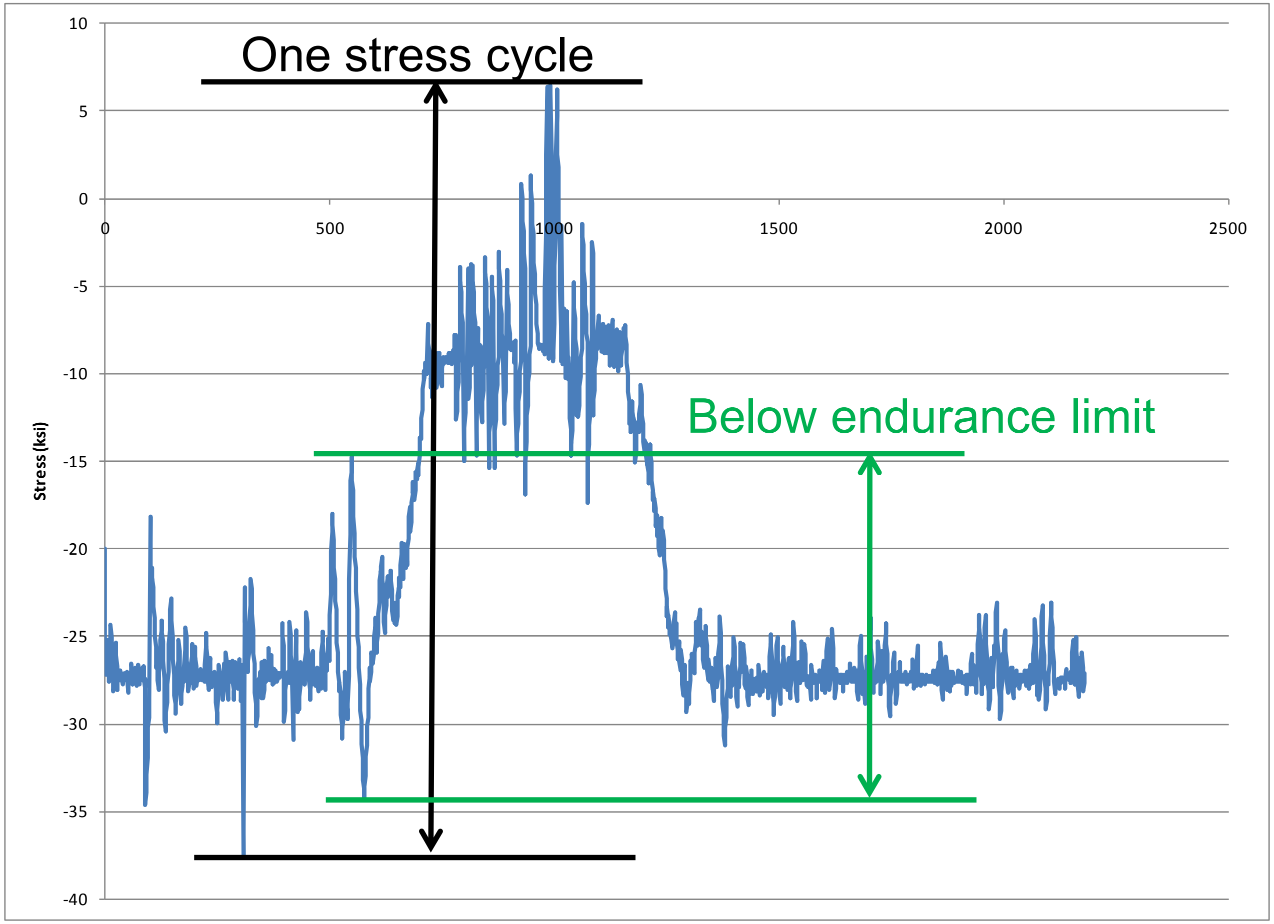
Stress Cycle Monitoring Example

Fact: Fatigue is path (order) dependent!



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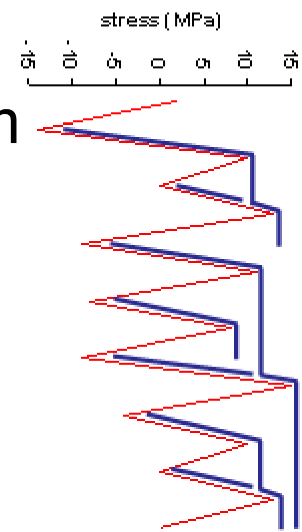
Counting Ordered Stress Cycles Using <u>Single Stress Term</u>

- Fatigue damage under random loading studied extensively in auto and aerospace industries.
- Several methods are available:
  - Range-pair
  - Rainflow
  - Ordered Overall Range (OOR)
- These different numerical methods produce essentially the same results.
- These methods are generally limited to single stress term using conventional algorithms.

Rainflow Algorithm

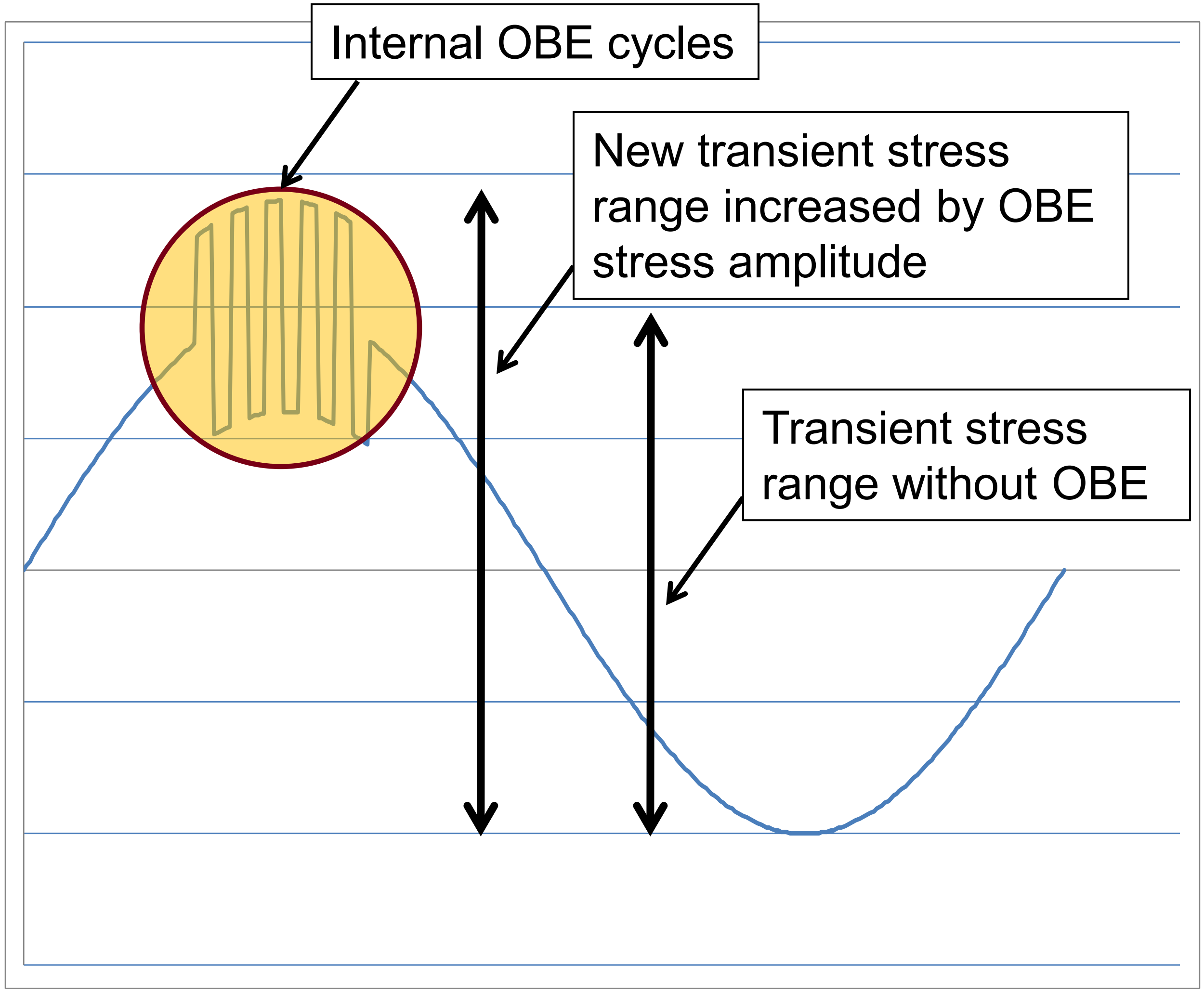
- Simplified Rainflow Cycle Counting Method documented in ASTM Standard No. E1049 (Reapproved 2005). *Standard practices for cycle counting in fatigue analysis*.
- ASME Section VIII Division 2 Annex 5.B (non-mandatory guidance).
- Peaks imagined as source of water that "drips" down a pagoda roof.
- Conventional algorithm uses single stress term
- Proportional loading



Order Dependence in ASME Code

ASME Code does not prohibit consideration of order. Methods for handling seismic events reflect order dependence.

Transient pair stress range increased by OBE stress amplitude. Remainder are internal (self) cycles.



The Multiaxial Challenge

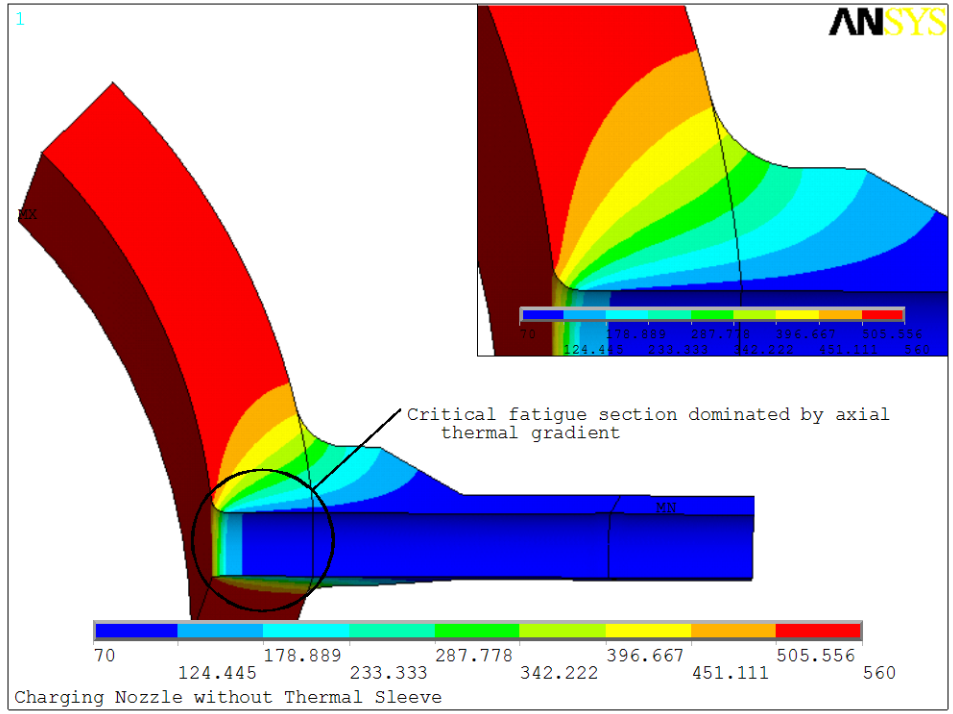
How to account for ordered stresses while meeting requirements and intent of ASME Code?

> *"In most cases it will be possible to choose at least one time during the cycle when the conditions are <u>known to be extreme</u>. In some cases it may be necessary to try different points in time to find the one which results in the largest value of alternating stress intensity."*

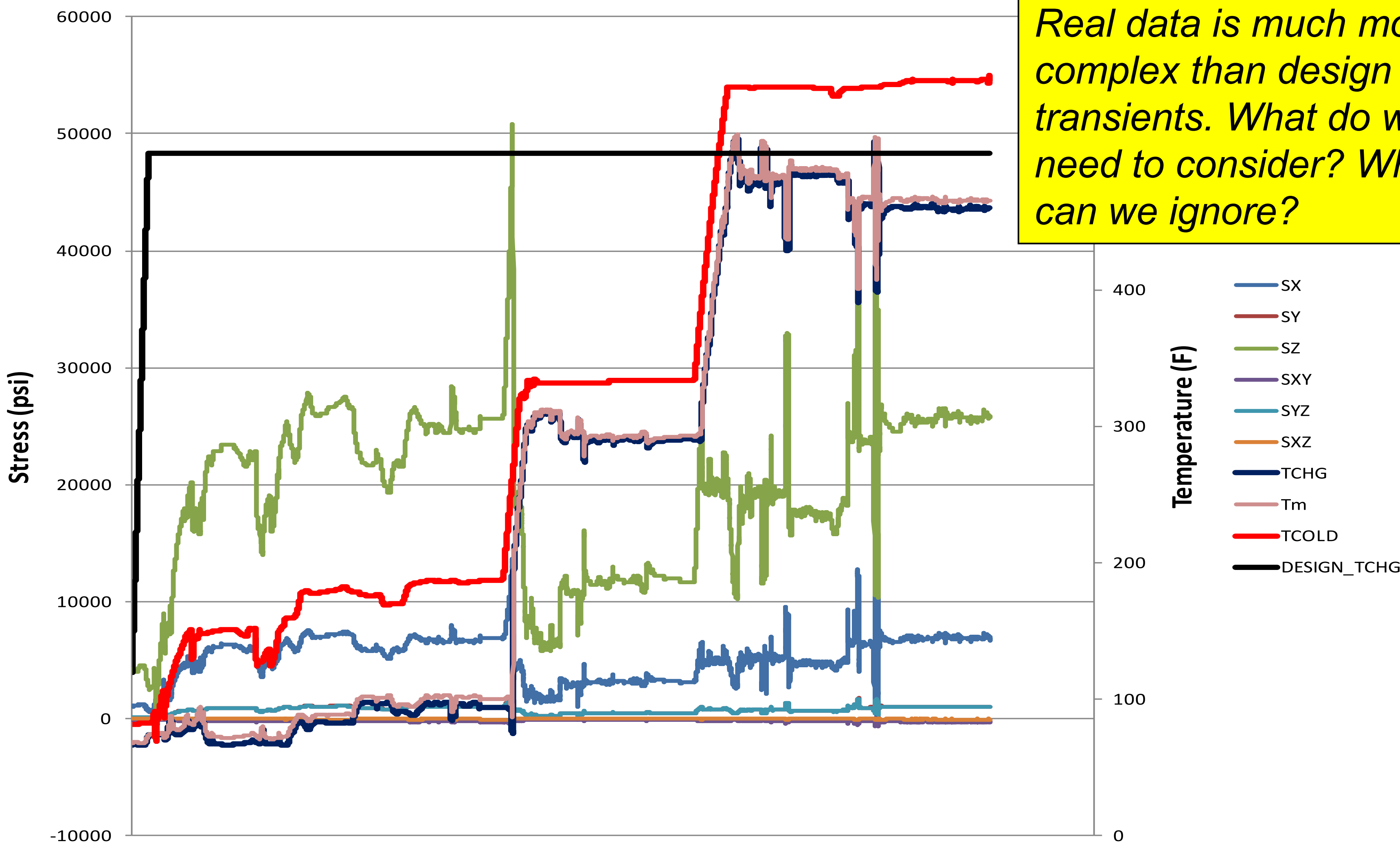
- NB-3216.2 provides guidance in computing stress intensity difference when normal and shear stresses may vary arbitrarily, and the "stress cube" that determines principal stresses may rotate.
- Challenge: SI between two points in a cycle is not equal to the stress intensity difference, which is determined based on the difference of the 6 individual stress components in going from one cycle to another.

Example - Charging Nozzle

- High steady state thermal gradient during cold injection.
- Difficulty with "design" type stress cycle pairing illustrated with complexity of real data.



Charging Nozzle – Plant Heatup



Real data is much more complex than design transients. What do we need to consider? What can we ignore?

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Solution Alternatives

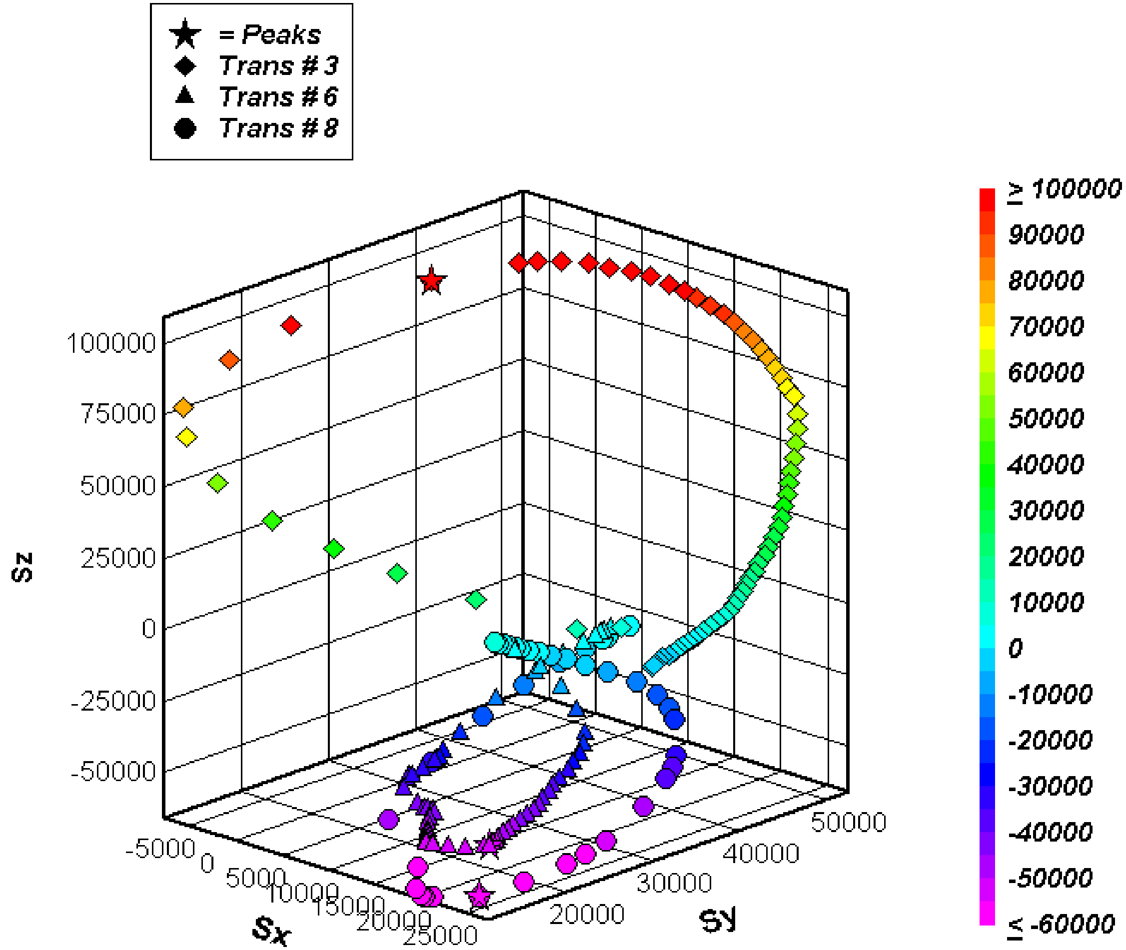
- A solution to the problem requires two important steps.
  - Multiaxial peak and valley detection logic
  - Multiaxial stress cycle pairing logic
- Several options investigated for each.
- Used together, “Rubberband” and “Rainflow-3D” produced best results across many test cases.

Criteria for Selection of Algorithm
How do we know what's right?

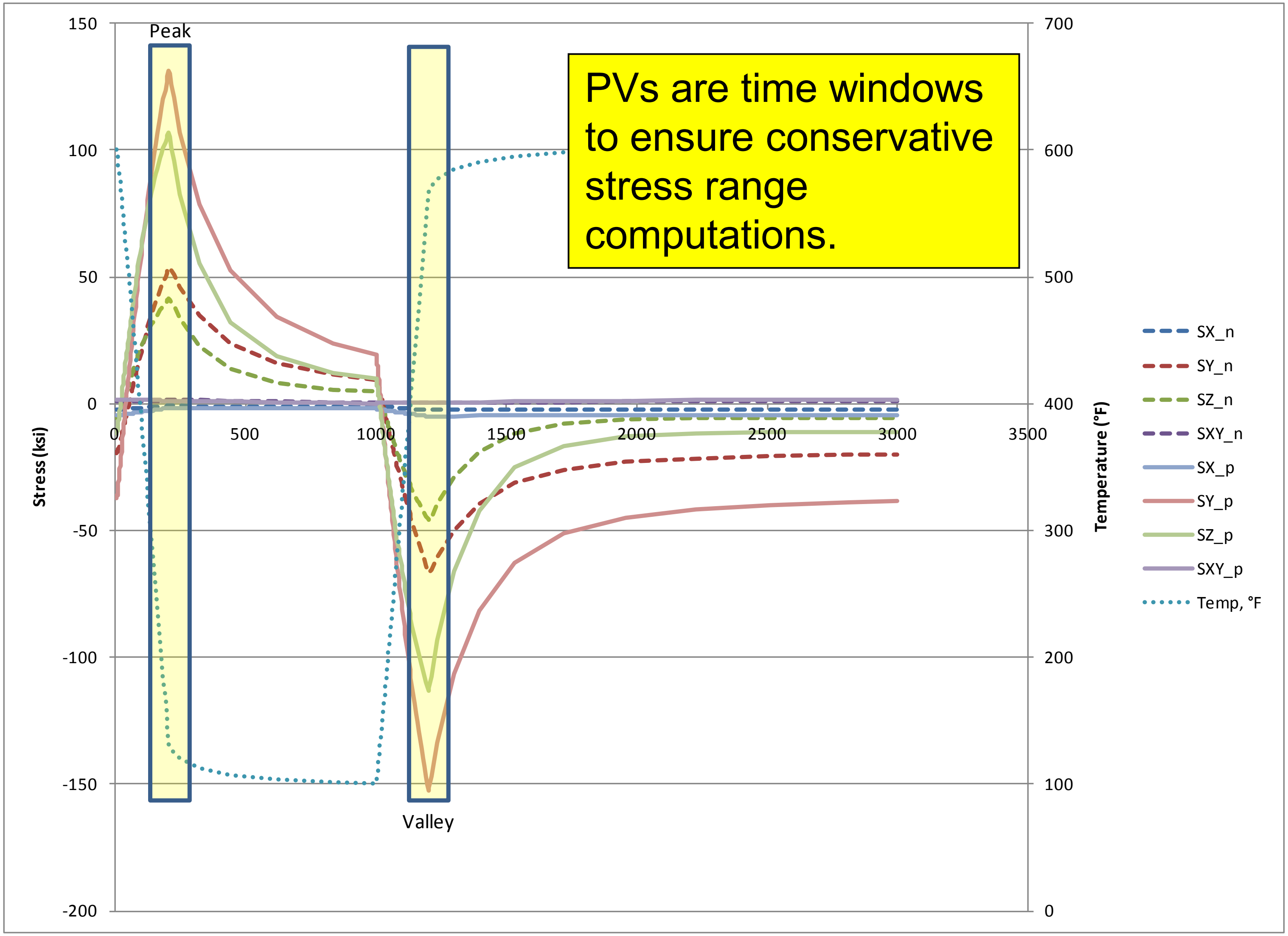
- When simulated in the same order, we should reproduce known problems from ASME NB-3200 design calculation examples (<u>benchmarks</u> for accuracy)
- Assuming a uniaxial stress with random, ordered loading, we should with our algorithms identify the same stress cycles as that from heavily vetted algorithms such as Rainflow (<u>validation</u> of sound engineering principles)
- Analyst judgments and manual adjustments to the stress cycle counting should not be necessary to produce consistently meaningful results (<u>repeatability</u> of results)

Peak/Valley Detection – "Rubberband"

- Detects points of maximum distance from the previous possible extrema.
- Looks for SI range to increase from previous extrema to a given threshold and then decrease to another threshold to identify a new extrema.
- Range varies in multiple dimensions.
- Filters out insignificant reversals well below the endurance limit
- Range of time included for each



Peak/Valley Detection – "Rubberband"



“Rubberband” Addresses General Concern in NRC RIS 2011–14 (December 29, 2011)

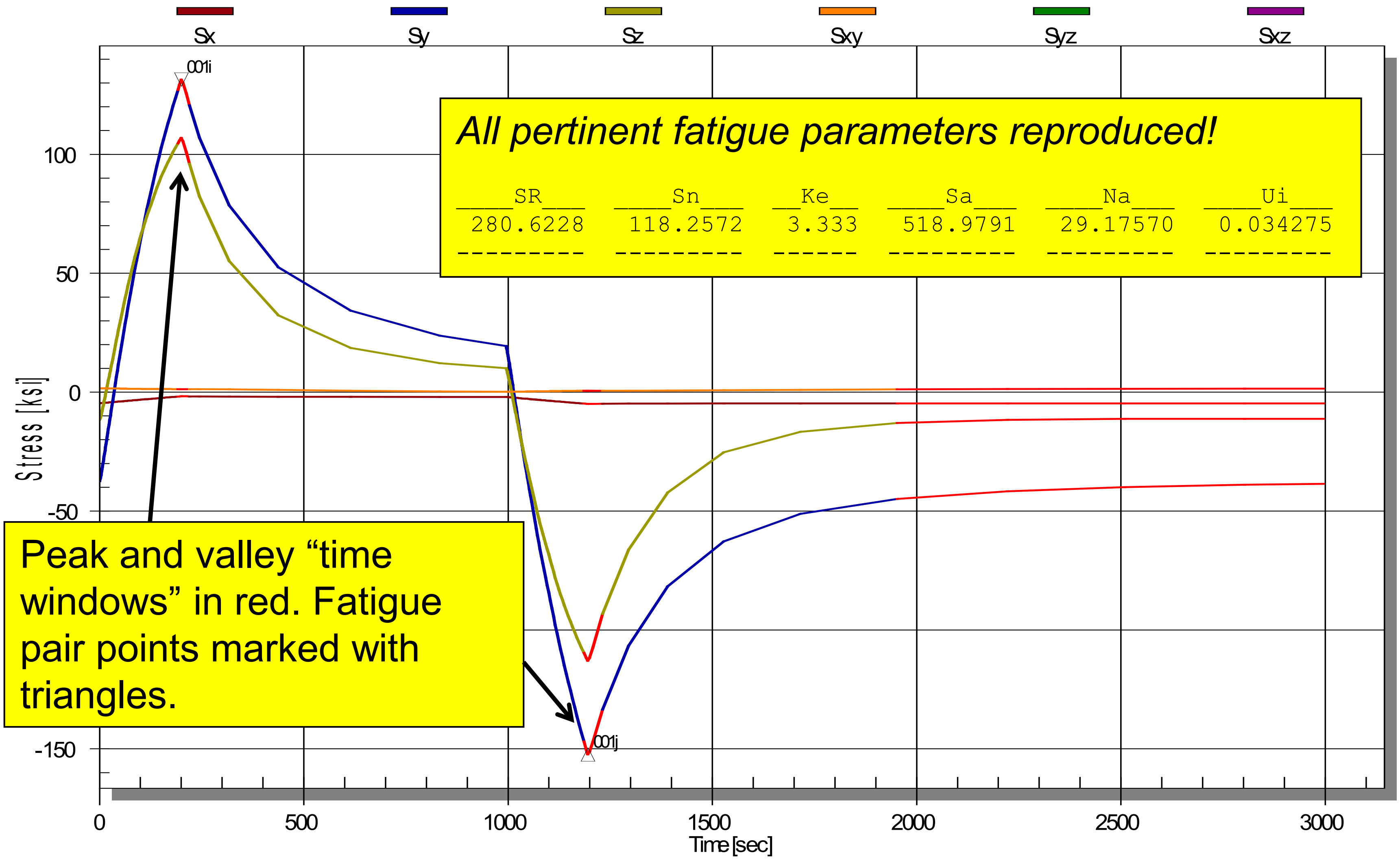
“Although this method of analyst intervention [*manual modification of peaks and valleys*] could provide acceptable results in some cases, reliance on the user’s engineering judgment and ability to modify peak and valley times/stresses, without control and documentation, could produce results that are not predictable, repeatable, or conservative.”

- By taking order and multiaxial stress range into account, manual peak and valley adjustment is not required.
- Process is predictable, repeatable and conservative.

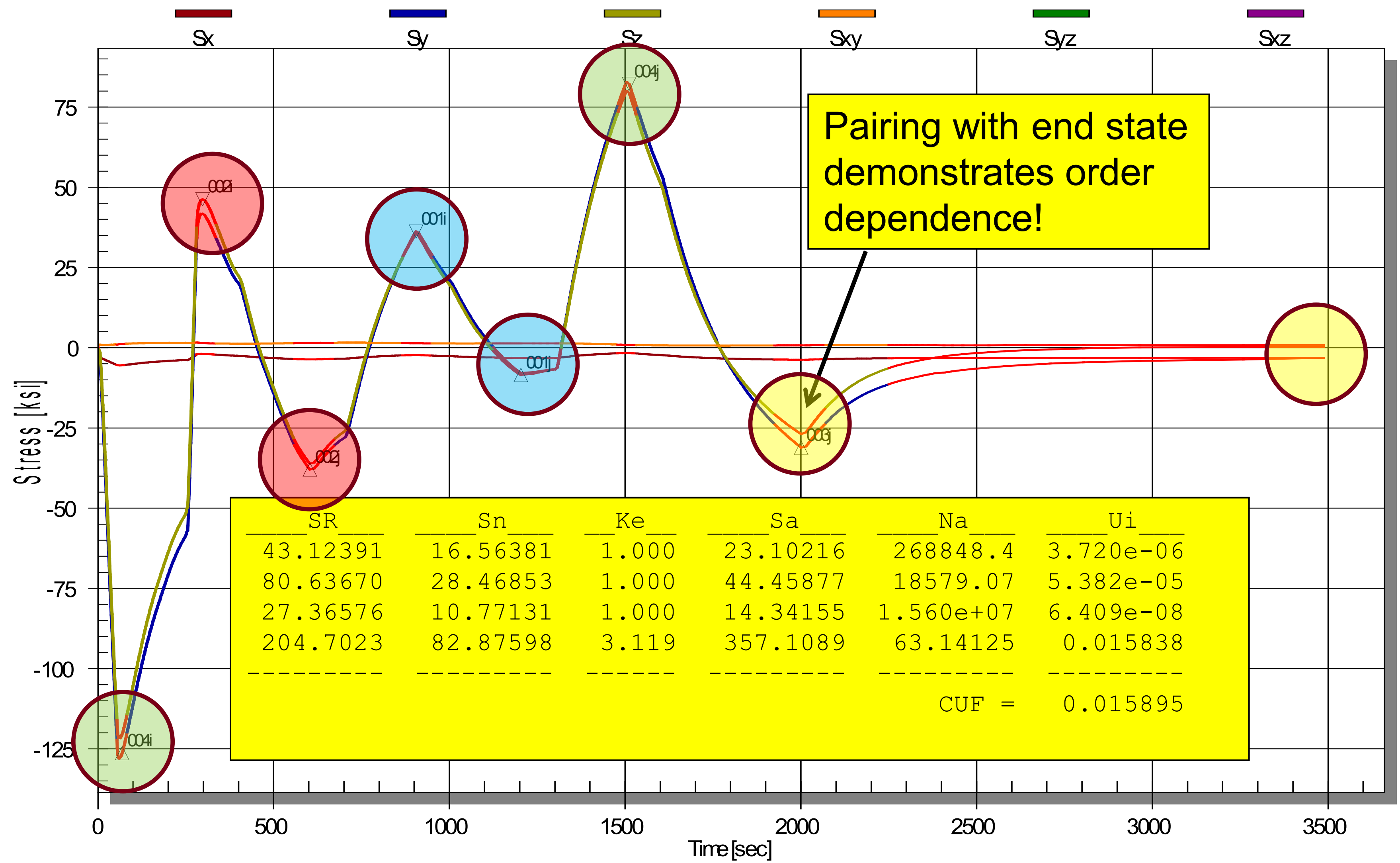
Pairing Logic – "Rainflow-3D"

- Conventional Rainflow algorithm implemented, but with following differences
  - SI range is computed between local extrema based on all six components of stress (instead of the algebraic difference in two values)
  - Each extrema contains at least one time point and likely represents a time window of more than one point.
  - Most conservative range pair selected based on combination of stress range, K_e (function of primary plus secondary stress intensity range) and the elastic modulus ratio.

Example – EAF Sample Problem Transient 1 – SCL 1



Operational Cycle with Multiple Stress Cycles



SR	Sn	Ke	Sa	Na	Ui
43.12391	16.56381	1.000	23.10216	268848.4	3.720e-06
80.63670	28.46853	1.000	44.45877	18579.07	5.382e-05
27.36576	10.77131	1.000	14.34155	1.560e+07	6.409e-08
204.7023	82.87598	3.119	357.1089	63.14125	0.015838
				CUF =	0.015895

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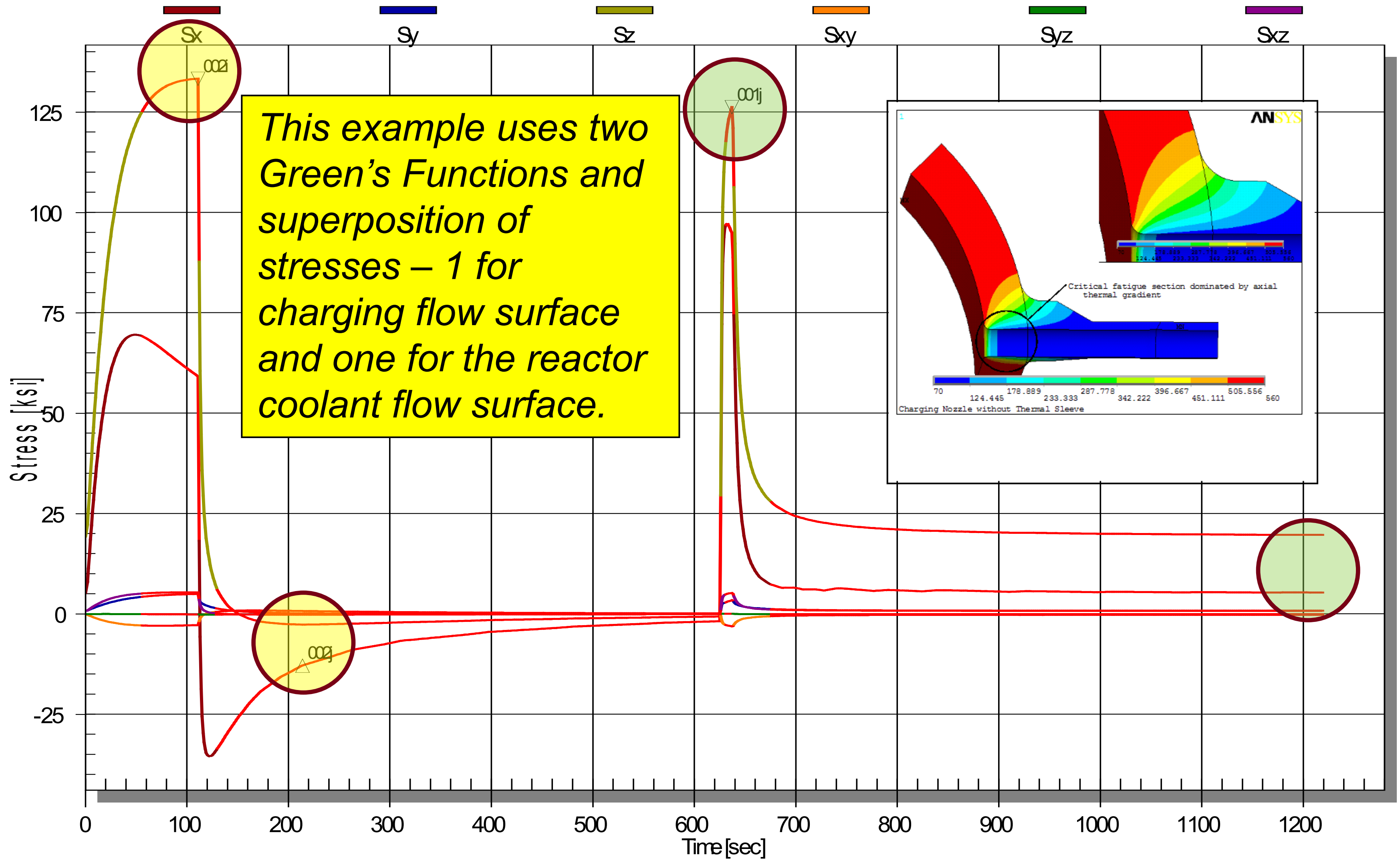
Charging Nozzle – Simulated Loss of Letdown with Delayed Return to Service



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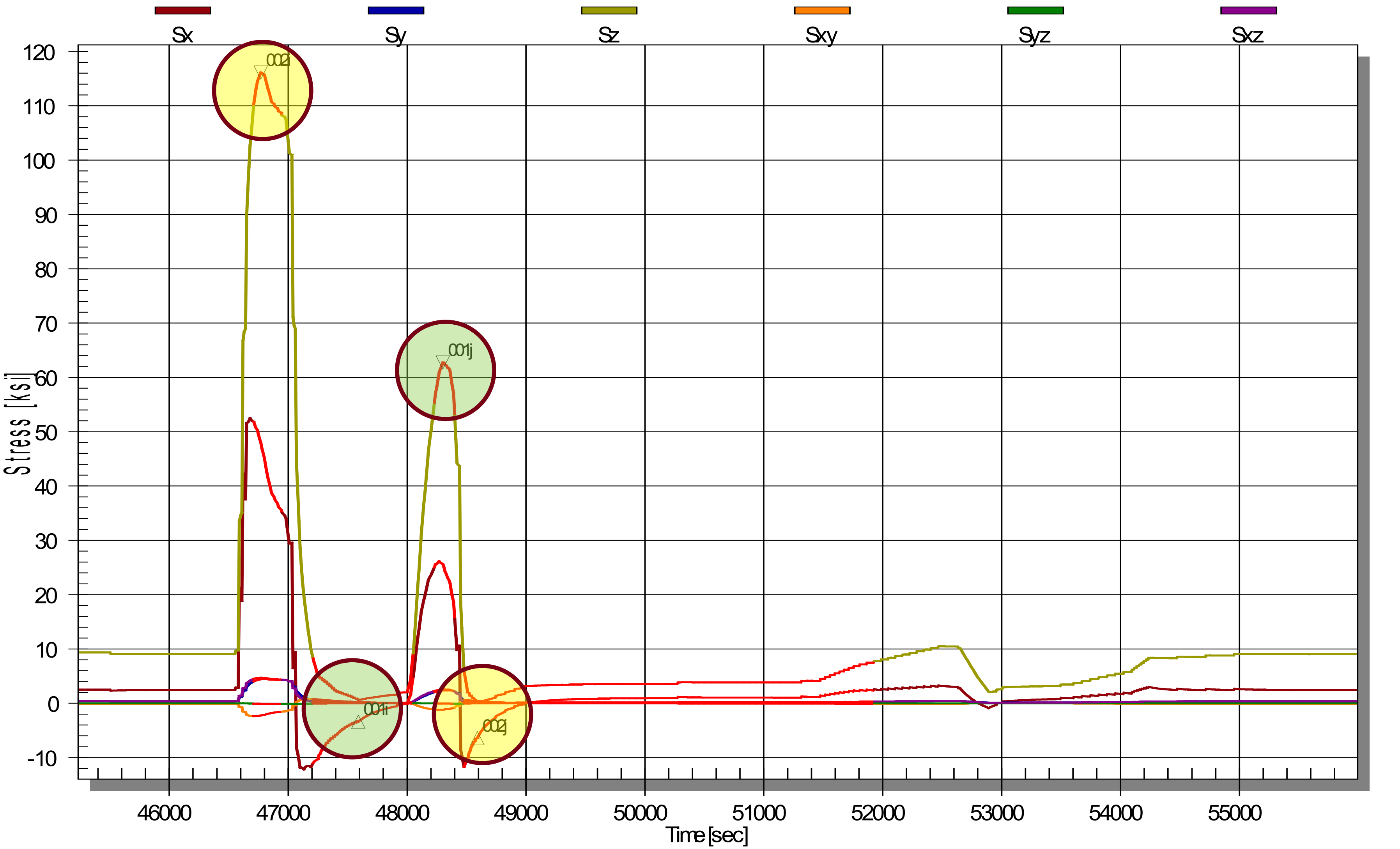
Charging Nozzle – Multiple Letdown Trips



Environmentally-Assisted Fatigue

- Methodology takes guidance from EPRI's EAF Expert Panel recommendations.
- Panel has reached general consensus on computation of strain rate using multiaxial stresses.
- Generally consistent with proposed Code case on strain rate (ASME Record No. 10-293) – ASME Code Case N-792-1

EAF Highlights

- Support rules of NUREG/CR-5704 / 6583 / 6909
- Computations at each time step:
  - Strain increment and strain rate
    - Auto determination of whether increasingly tensile or compressive, based on largest absolute value principal stress of the stress differences
    - Possible inclusion of K_e strain rate can reduce conservatism > 25% (not currently allowed in MRP-47 Rev. 1 or Japanese Code)
  - F_{en} as a function of:
    - Current service temperature
    - Computed strain rate
    - Dissolved oxygen level via user-input time history or direct instrumentation
    - Other user inputs (sulfur content, etc.)

$$F_{en} = \frac{\sum F_{en,i} \Delta \varepsilon_i}{\sum \Delta \varepsilon_i}$$

- F_{en} for each stress cycle
  - Integration (modified rate) approach

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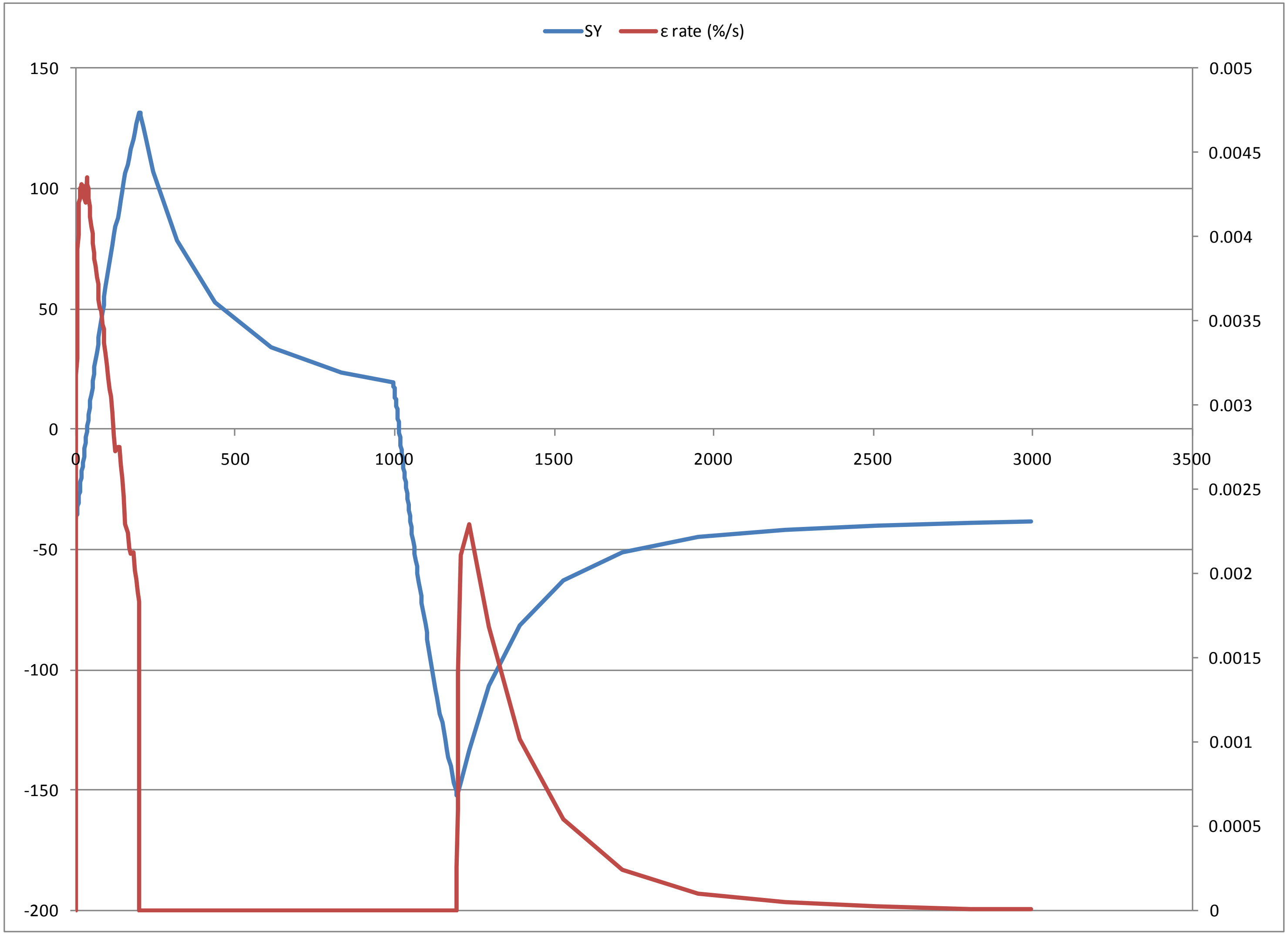
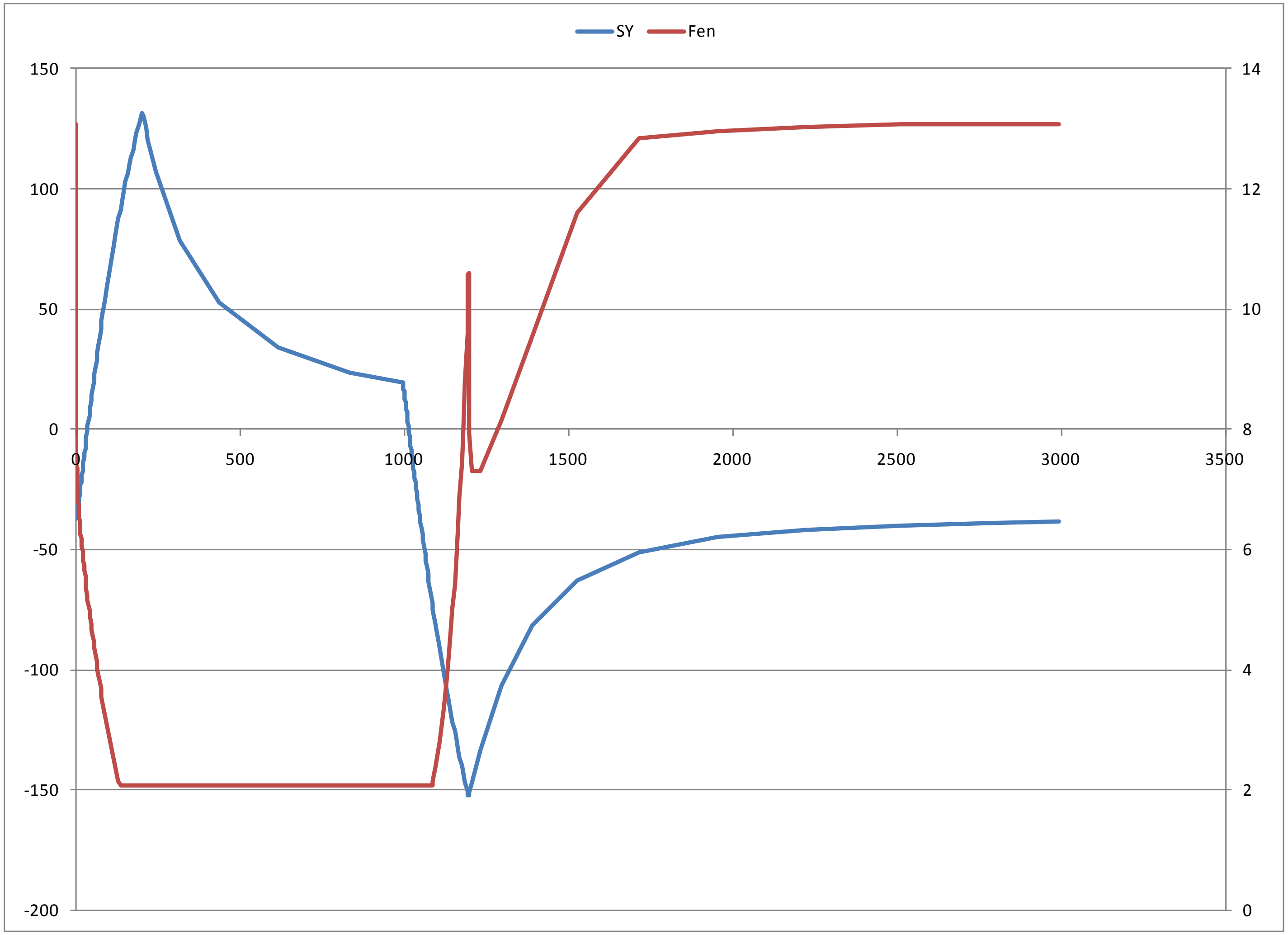
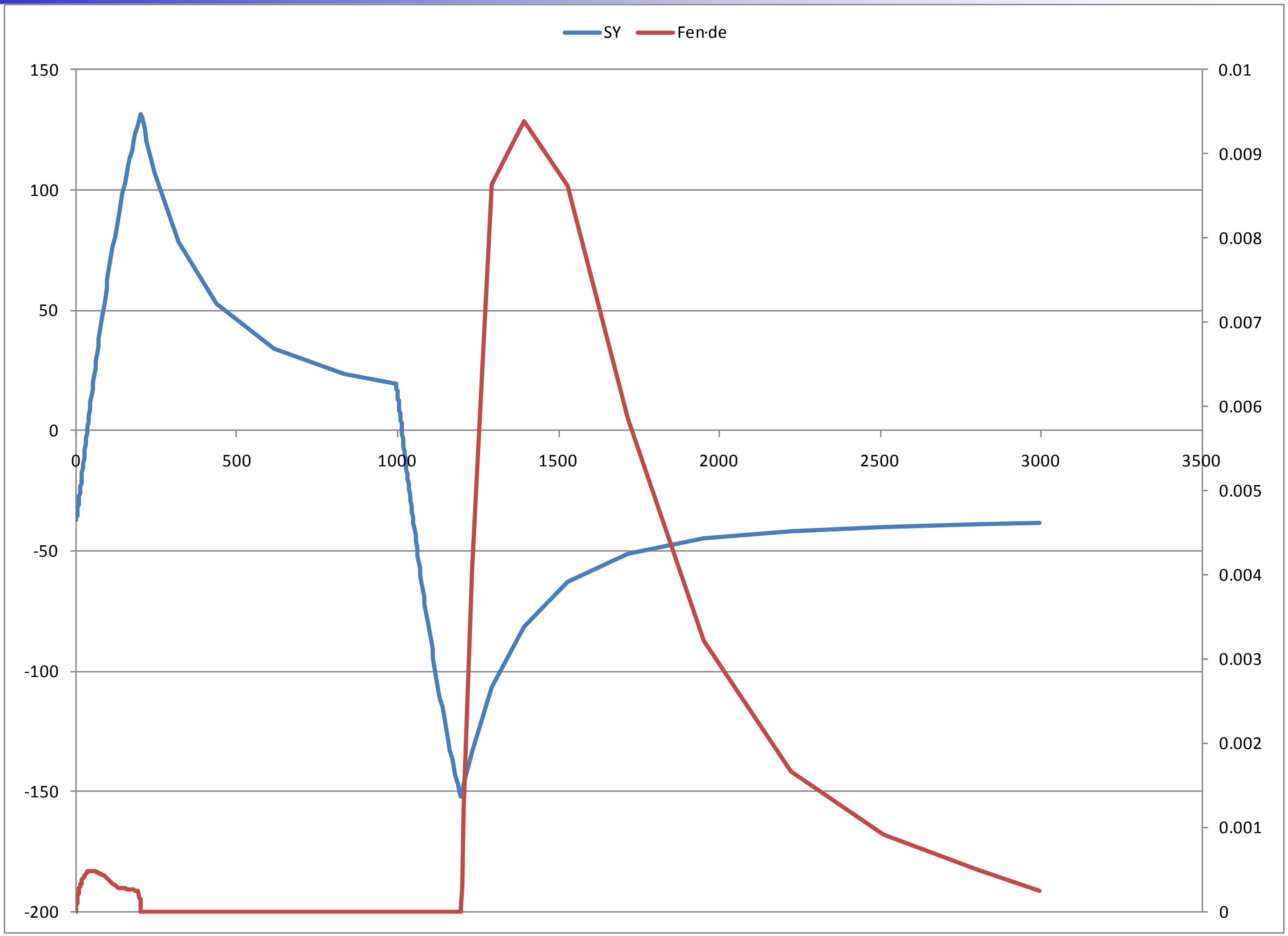
EAF Sample Problem Calculation



EAF Sample Problem Calculation



EAF Sample Problem Calculation



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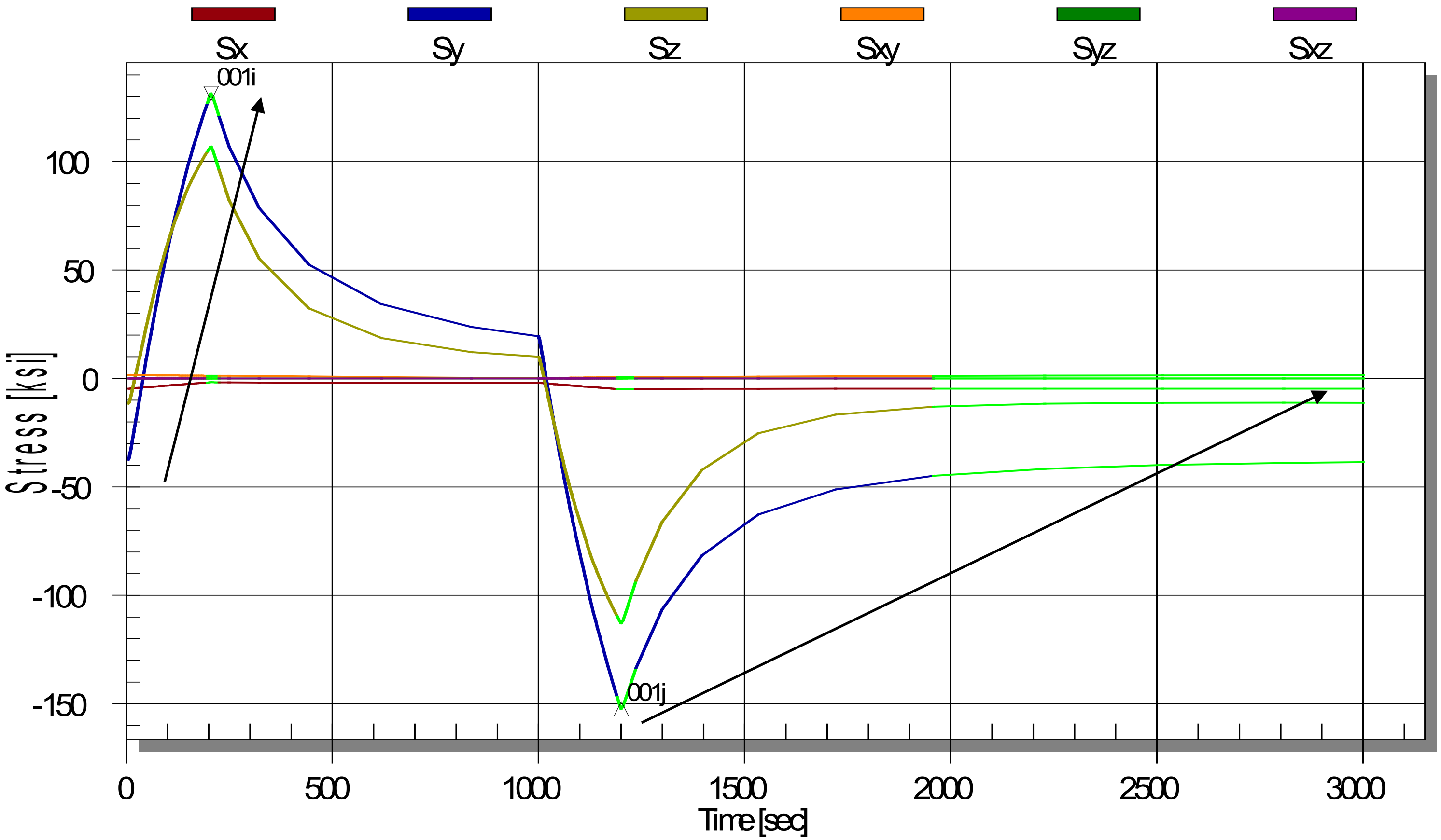
Bounds of F_{en} Integrations

- Integrates from Valley to its adjacent Peak and to Peak from its adjacent Valley



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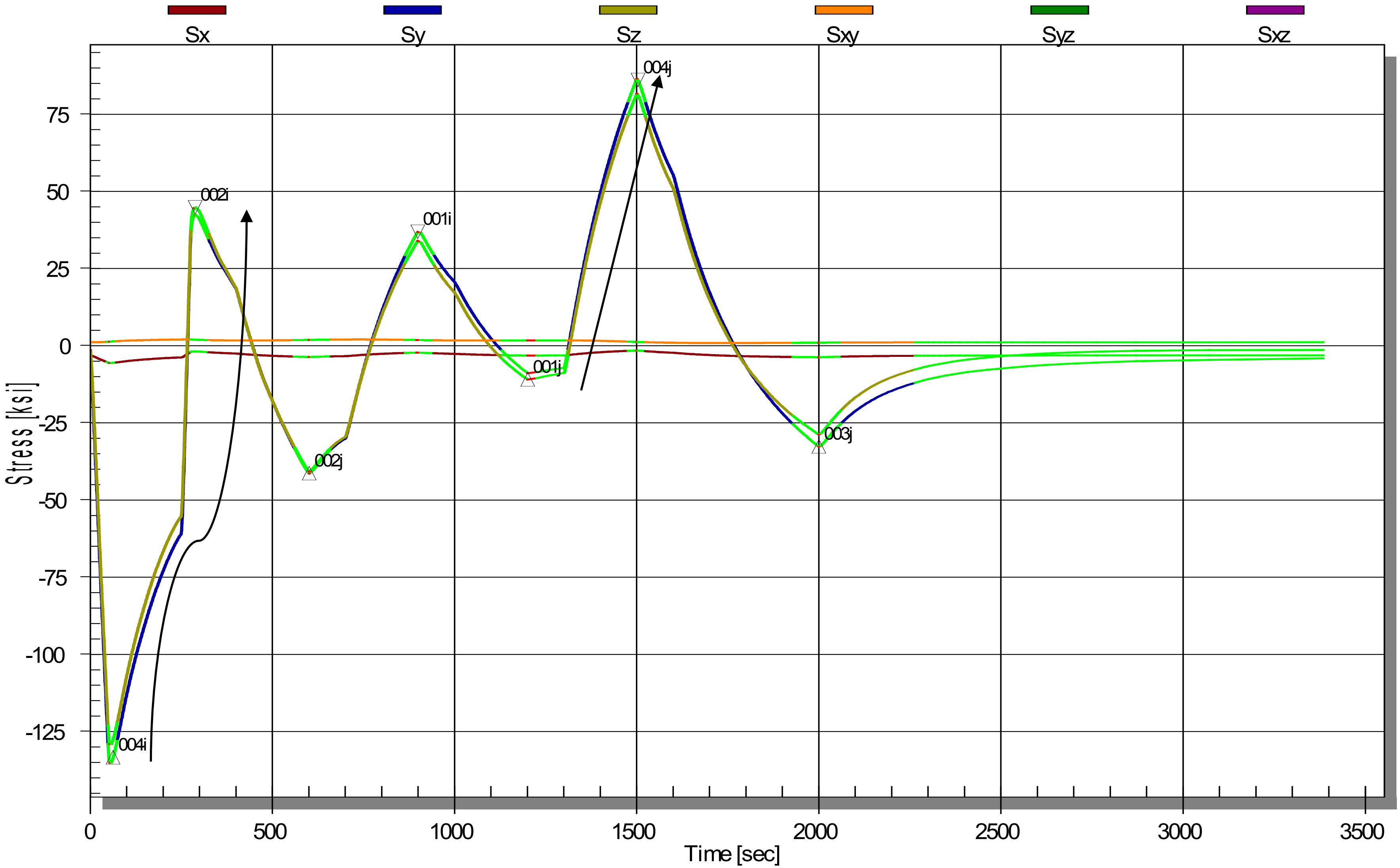
F_{en}'s for Complex Stress Cycling



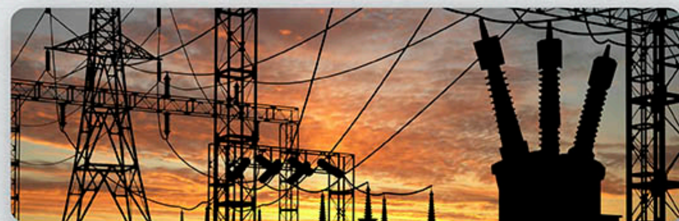
Conclusions

- Overall methodology combines many proven practices
- Basic steps in the process include:
  - Multiaxial stress calculations
    - Address NRC RIS 2008-30
    - Accurate knowledge of through-wall distributions
  - Smart Peak/Valley Detection and Stress Cycle Counting
    - Rubberband (detects reversal regions using multiaxial stress range criteria)
    - Rainflow-3D (identifies stress cycles)
  - Calculation of EAF
    - Meets GALL requirements
    - Implements Expert Panel guidance

Questions and Comments

Attachment 5

IMPROVED BASIS AND REQUIREMENTS FOR BREAK LOCATION POSTULATION PRESENTATION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Improved Basis and Requirements for Break Location Postulation

EPRI Technical Report 1022873, October, 2011

Terry J. Herrmann, P.E.
Senior Associate

NRC Fatigue Meeting
January 5, 2012

Presentation Objective

- Provide a basis for revising NUREG-0800, BTP 3-4 criterion for postulating High Energy Line Break (HELB) locations.
- In the interim, allow utilities to use the alternate approach presented here to the existing fatigue usage criterion for postulating HELB locations on a case-by-case basis.

Introduction/Background

- Currently, for plants with piping systems designed to ASME III, a cumulative usage factor (CUF) of 0.1 is a criterion for postulating break locations in reactor coolant pressure boundary piping (NUREG-0800, BTP 3-4).
  - Continued use of the 0.1 CUF criterion may result in additional costs without any risk benefit for new plants and plants pursuing extended operation.
  - No clear technical basis exists for this value.
  - The original objective was to provide margin to the Code limit of 1.0 to account for uncertainties.
  - Over 4 decades of industry experience demonstrates that large leaks from fatigue damage does not occur from design transients used in CUF calculations.
  - A number of damage mechanisms have been identified and dispositioned since the CUF criterion was promulgated.

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Introduction/Background

- A risk-informed approach would provide a technical basis for revising the current fatigue criterion, consistent with the NRC's Risk-Informed and Performance-Based Plan (RPP) initiative.
- Rupture probability, combined with consequences would be a good measure for assessing the risk of postulated breaks.
- Leak probability is suggested as a surrogate for rupture probability
  - More straightforward to estimate than rupture probability.
  - Computation less controversial.
  - More conservative, overall.

Introduction/Background

- EPRI contracted with Structural Integrity Associates (SI) to explore break location postulation criteria other than CUF=0.1.
- Industry operating experience insights are summarized.
- The results of analyses to explore leak probability and fatigue usage factors for a selection of components are also presented.
  - CUF without consideration of environment.
  - CUF_{en} (CUF considering environmental influence on fatigue).

Insights from Industry Operating Experience

- Available sources of piping system failures (NRC, EPRI, SI, and others) were reviewed.
  - Over 4,900 worldwide events were collected representing over 9,000 reactor critical years between 1970 and 2005.
- Data was lacking to quantitatively relate design CUF to failures in service.
- Less than 5% of piping cracks, leaks or ruptures were associated with thermal fatigue.
- Several sources noted that the majority of these failures were associated with a few well-documented generic issues which had not been anticipated during design (next slide).

Insights from Industry Operating Experience

- Some of the damage mechanisms that have been identified and dispositioned through improved regulatory guidance since the CUF criterion was promulgated include:
  - BWR feedwater and CRD nozzle cracking (NUREG-0619).
  - Feedwater piping cracking in PWRs (Bulletin 79-13).
  - Stagnant borated water systems (Bulletin 79-17).
  - Intergranular stress corrosion cracking (Generic Letter 88-01).
  - Leakage at valves (Bulletin 88-08).
  - Thermal stratification (Bulletin 88-11).
  - Erosion/corrosion (Generic Letter 89-08).
  - Reactor water environmental effects on fatigue (NUREG/CR-6909 and other documents identified in NUREG-1801).

Methodology & Analytical Approach

- Methodology based on NUREG/CR-6674 Fatigue Analysis of Components for 60-Year Plant Life):
  - Consistent with prior studies.
  - Considers environmental effects on fatigue usage.
  - Considers Impact on Core Damage Frequency.
  - Addresses a range of fatigue-sensitive locations for each plant design as well as newer and older vintage plants.

Methodology & Analytical Approach

- Information used from NUREG/CR-6674:
  - Estimated fatigue stresses and cycles.
  - Reactor water environmental parameters (strain rate, oxygen content and temperature).
  - Leak probabilities and cumulative usage factors.
  - Leak probabilities use pcPRAISE with older probabilistic strain-life relations (NUREG/CR-6335).

Methodology & Analytical Approach

- Updated ANL strain-life relationships from NUREG/CR-6909 were used to address environmental effects.
  - pc-PRAISE used to calculate leak probabilities vs. operating time using cyclic stresses and environments from NUREG/CR-6674.
  - CUF_{en} computed using new ASME design fatigue curve including environment.
  - Convert operating time to CUF_{en} assuming linear fatigue damage accumulation with time.
  - Evaluate core damage frequency using information from NUREG/CR-6674 [P (core damage)|rupture]
  - Plot Core Damage Frequency (CDF) vs. CUF_{en}
  - Compare CDF to PSA Applications Guide (TR-105396) Criteria

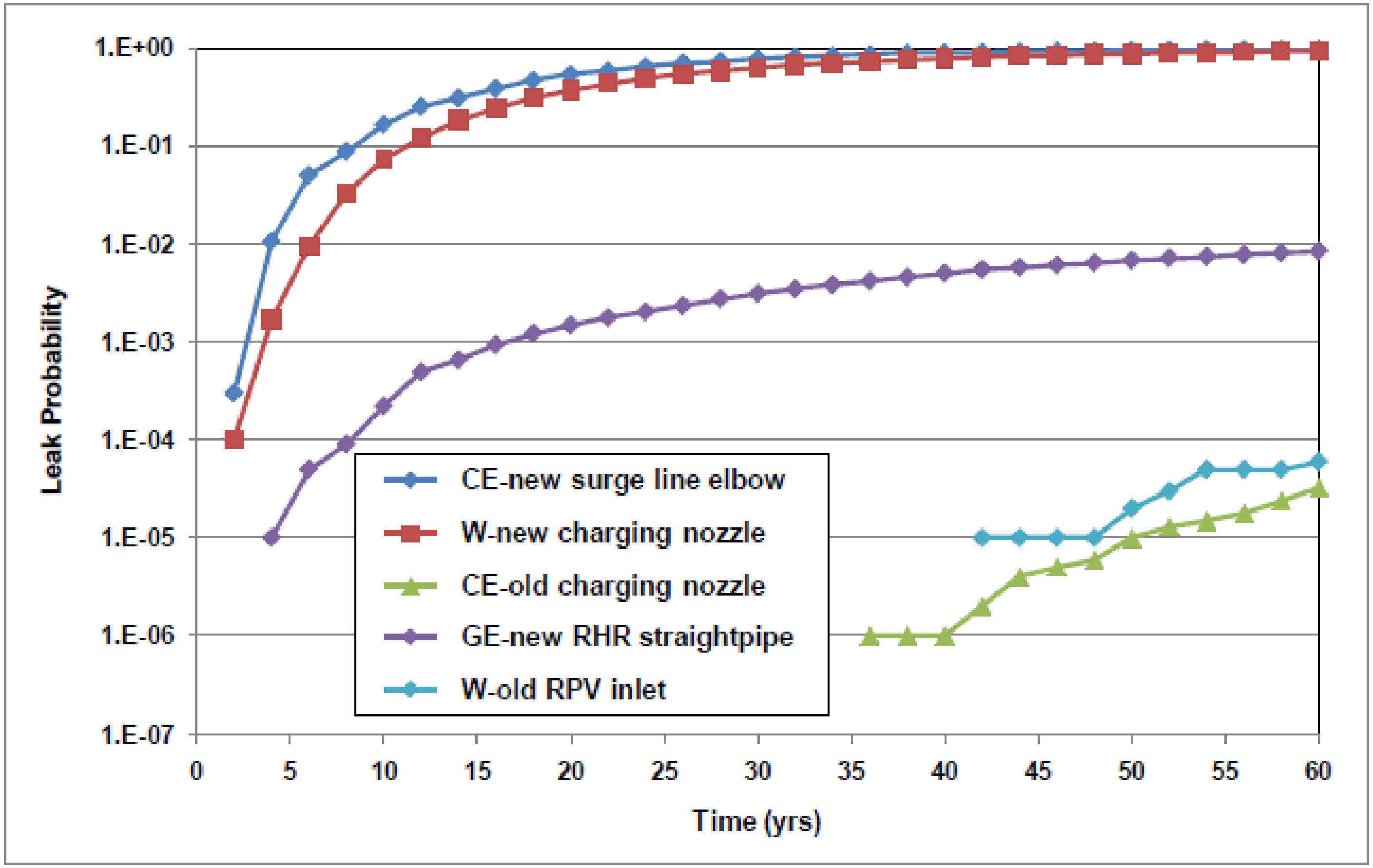
Component Selection

5 components were selected from NUREG/CR-6674, considering material type, cumulative usage and environment (env/air is ratio of fatigue usage factor based on a comparison of air and reactor water results from NUREG/CR-6674).

#	Name	NUREG/ CR-6260 Section	matl	*CUFen(60)*	*Env/air*	$P_{lk}(60)$	Comment
4	CE-new surge line elbow	5.1.3	SS	3.90	2.65	0.998	high failure prob.
24	W-new charging nozzle	5.4.4	SS	5.06	4.08	0.963	
14	CE-old charging nozzle	5.2.4	SS	0.843	2.11	6.0×10^{-4}	low CUF, low env. factor
39	GE-new RHR straightpipe	5.6.6	LAS	16.9	27.66	0.621	high CUF, high env. factor
28	W-old RPV inlet	5.5.2	LAS	0.453	2.23	0.0504	low CUf, low env. factor

Cumulative Leak Probability for Selected Components

pc-PRAISE leak probability calculations were based on NUREG/CR-6909. Cumulative leak probability results are plotted below.



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Core Damage Frequency Estimation

Core Damage Frequency is related to leak frequency using the following information from NUREG/CR-6674.

#	name	matl	*P(CD | leak)*
4	CE-new surge line elbow	SS	2.85×10^{-5}
24	W-new charging nozzle	SS	8.00×10^{-6}
14	CE-old charging nozzle	SS	8.00×10^{-5}
39	GE-new RHR straightpipe	LAS	9.02×10^{-9}
28	W-old RPV inlet	LAS	2.70×10^{-6}

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Core Damage Frequency Estimation

- pc-PRAISE results two slides earlier show the cumulative distribution function for the leak probabilities.
- These are converted to leak frequencies by taking the slope of the curve *dPlk(t)/dt*.
- Leak Frequency is converted to Core Damage Frequency by multiplying by P(CD|leak) from the previous slide.
- Operating time is converted to CUF_{en} using NUREG/CR-6909 fatigue design curves with linear fatigue damage accumulation with time.

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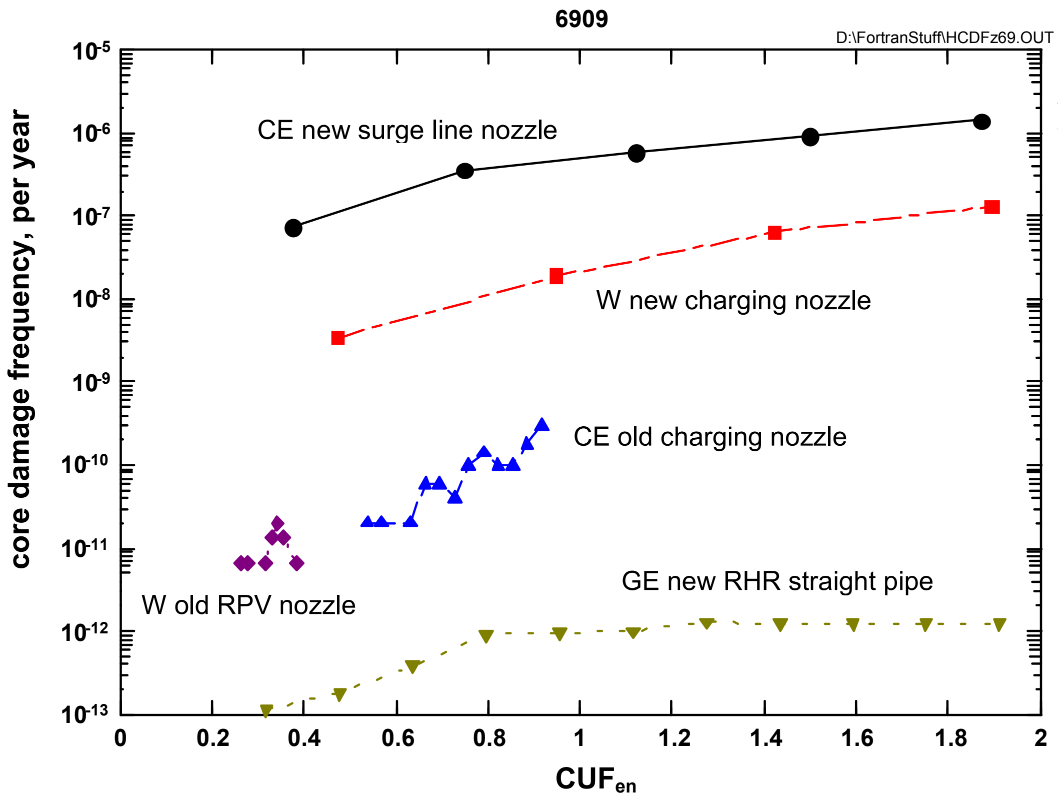
Core Damage Frequency vs. CUF_{en} Results



The lack of a direct correlation between Core Damage Frequency and component CUF_{en} values compromises efforts to use specific values of CUF_{en} as a criterion for postulating HELB locations.

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No Direct CUF_{en} Correlation with Leak – Why?

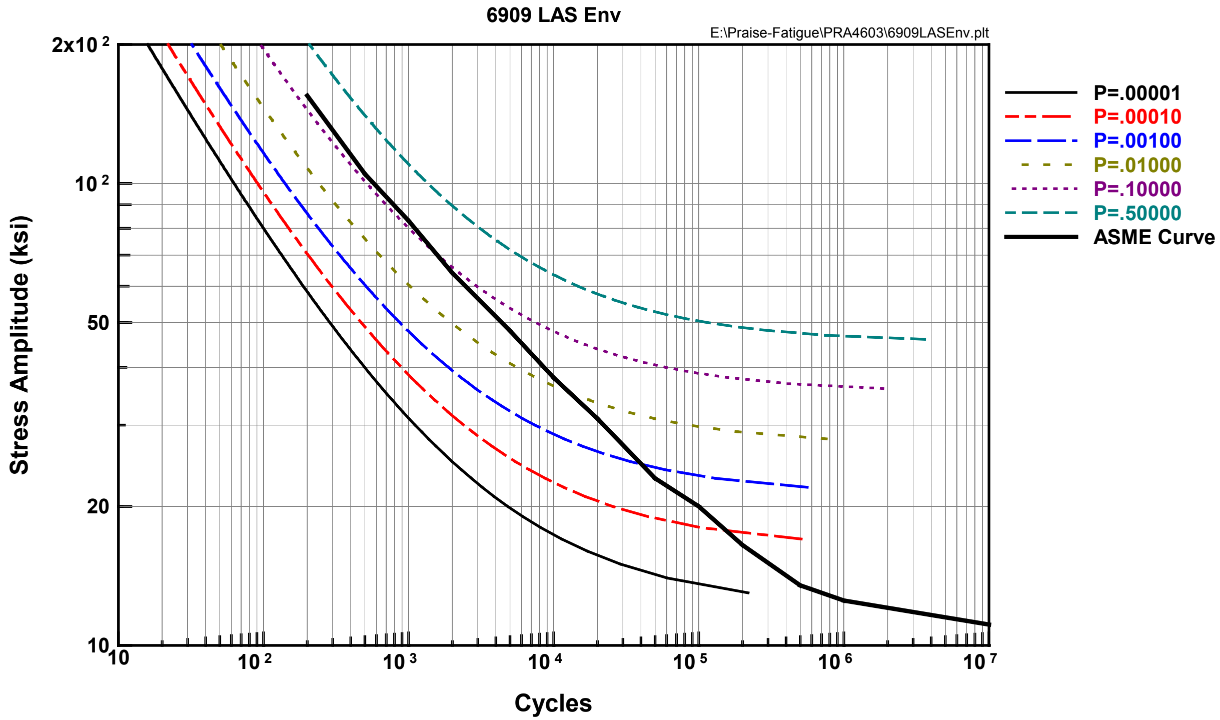
- Intuitively CUF_{en} should correlate well with initiation / leak probability.
- There are many factors that compromise the relationship between leak frequency and ASME calculated CUF_{en} factors. These factors include:
  - Stress profile (membrane, bending, radial gradient thermal)
  - Geometry (use of stress indices)
  - CUF methodology (strain-life correlations)
  - Stress evaluation methods (NB-3600 vs. NB-3200)
  - Crack growth considerations
  - Material, temperature
  - Crack growth relationships

No CDF vs. CUF_{en} Correlation – Why?

- Earlier slide shows no direct correlation between component CUF_{en} values and Core Damage Frequency among components evaluated.
- Probabilistic initiation data vs. ASME design curves
  - ASME design curves do not follow a line of constant crack initiation probability.
- Probability of CDF given leakage varies for different components.
  - Even if there was a good correlation between leak probability and CUF_{en}, agreement would be eliminated by component-specific P(CD|leak) relationships.

No CDF vs. CUF_{en} Correlation – Why?

The ASME Design curve is not consistent with initiation probability fractiles based on statistical analysis of fatigue data used in pc-PRAISE initiation and leak probability calculations.



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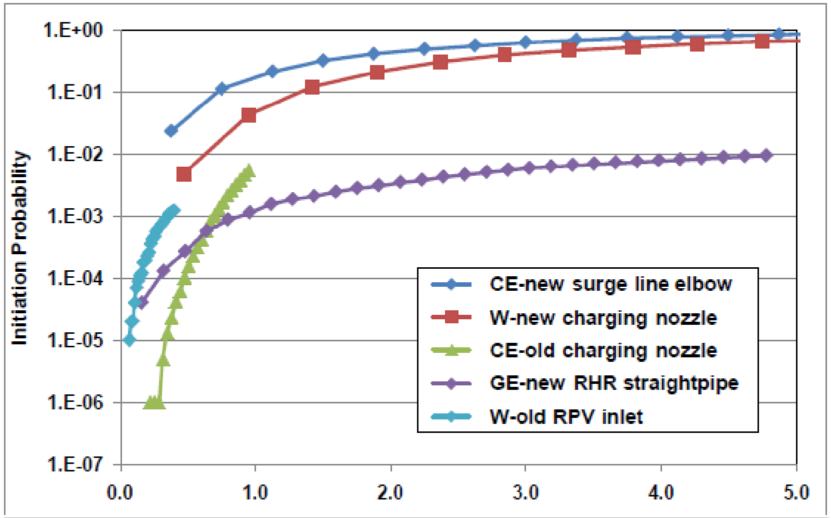
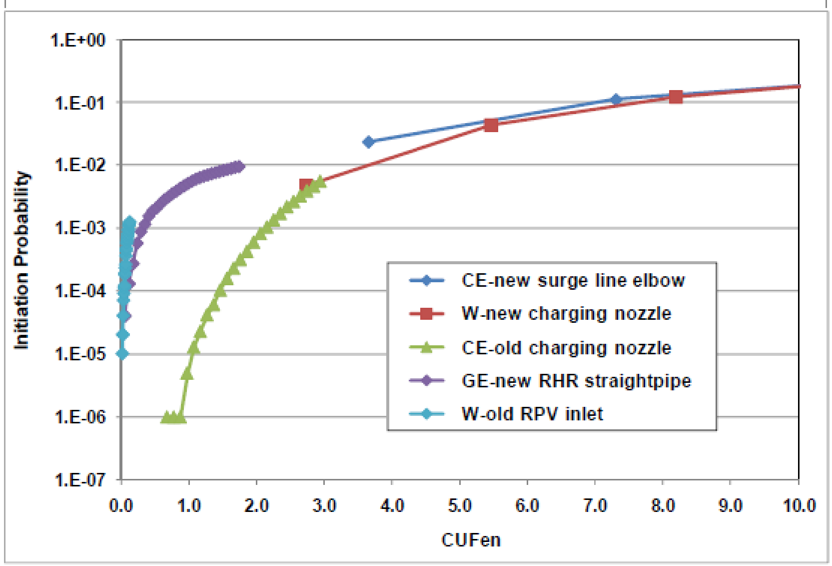
Initiation Probability Comparison of ASME Design Curve vs. Fatigue Data Fractiles





- Top plot shows initiation probabilities plotted versus to CUF_{en} factors calculated using the ASME design curve.
- No direct correlation observed due to inconsistency between fatigue initiation fractiles and the ASME design curve.

- Bottom plot shows initiation probabilities plotted versus CUF_{en} factors calculated using the 0.1% fractile of the fatigue data.
- Better correlation observed.
- SS and LAS separate out.

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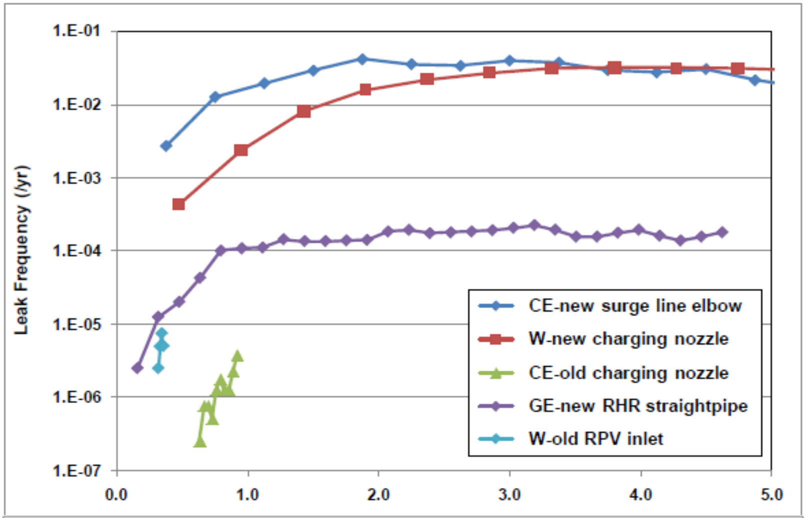
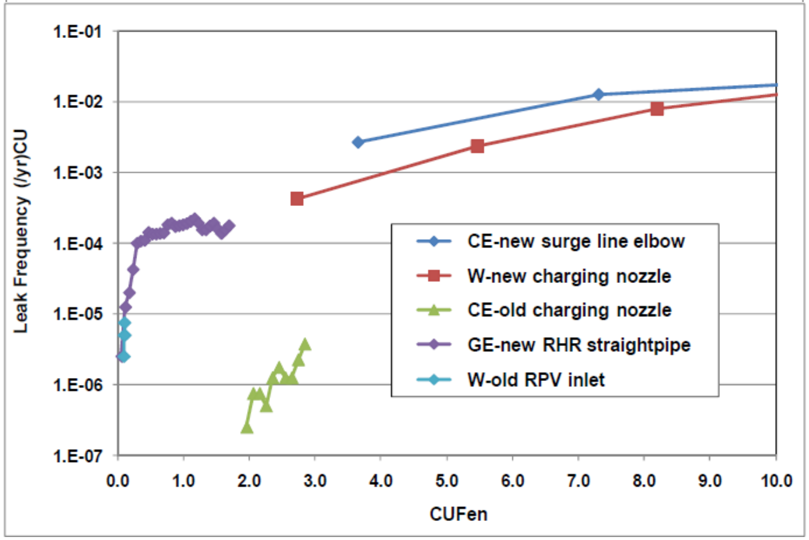
Leak Probability Comparison of ASME Design Curve vs. Fatigue Data Fractiles





- Top plot shows influence of using ASME design curve on the relationship between leak frequency and component CUF_{en}.
- Poor correlation observed when using ASME design curve.

- Bottom plot shows leak frequencies versus CUF_{en} calculated using the 0.1% fractile of the fatigue data.
- Comparison to corresponding initiation probabilities shows influence of geometry, spatial stress gradients, KIC and da/dN material relationships.

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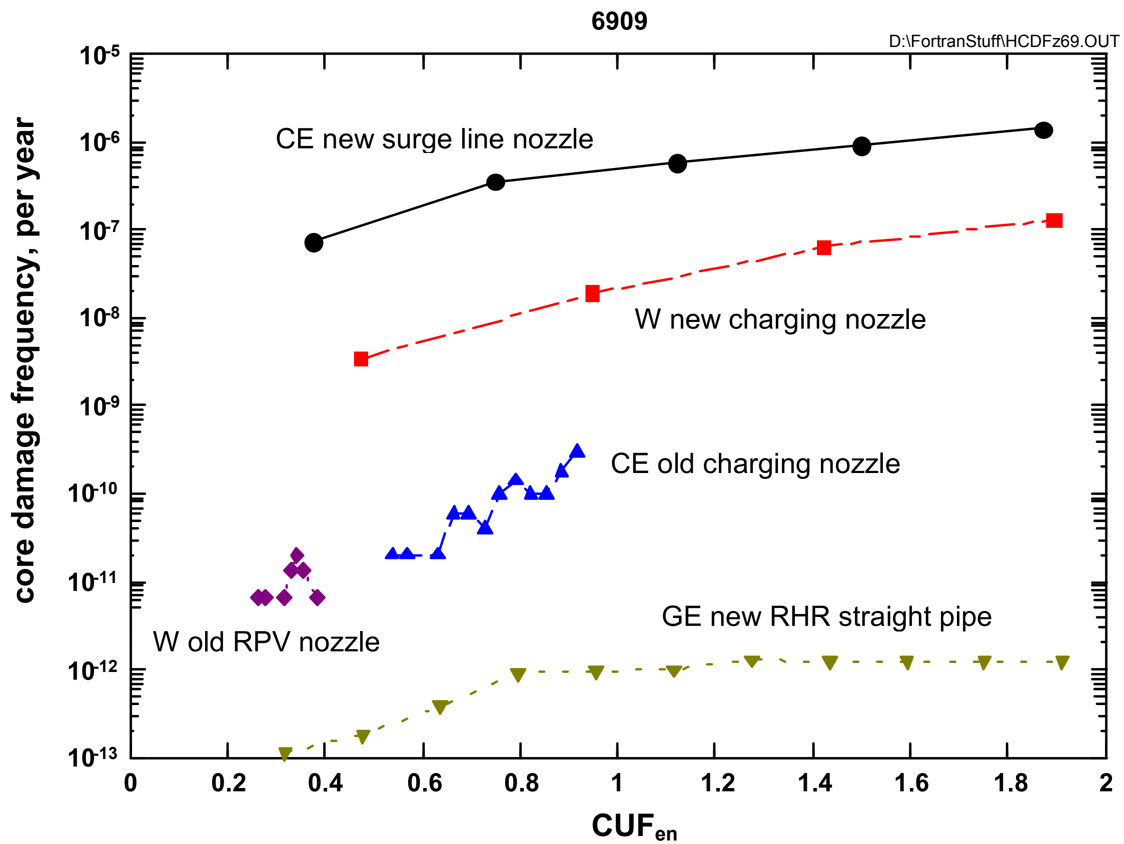
What Fatigue Criterion Would be Appropriate?

- From the previous slides, it is clear that evaluating the change in CUF_{en} criterion from that of 0.1 to some other value is significantly hampered due to inconsistencies in the impact of CUF_{en} on initiation, leak probability, and CDF.
- Therefore, the impact of an CUF_{en} value of 1.0 was evaluated, consistent with the ASME Code and what is considered to be acceptable for other plant locations, per NUREG-1801.
- The risk associated with the design of the plant is compared to the NRC's CDF goal of less than $1x10^{-4}$/year promulgated in NUREG-0800 Chapter 19 and to Regulatory Guide 1.174 for changes from baseline CDF values.

Core Damage Frequency vs. CUF_{en} Results



The above plot shows that an CUF_{en} of 1 results in a CDF less than $1x10^{-6}$ in all cases for the selected components.

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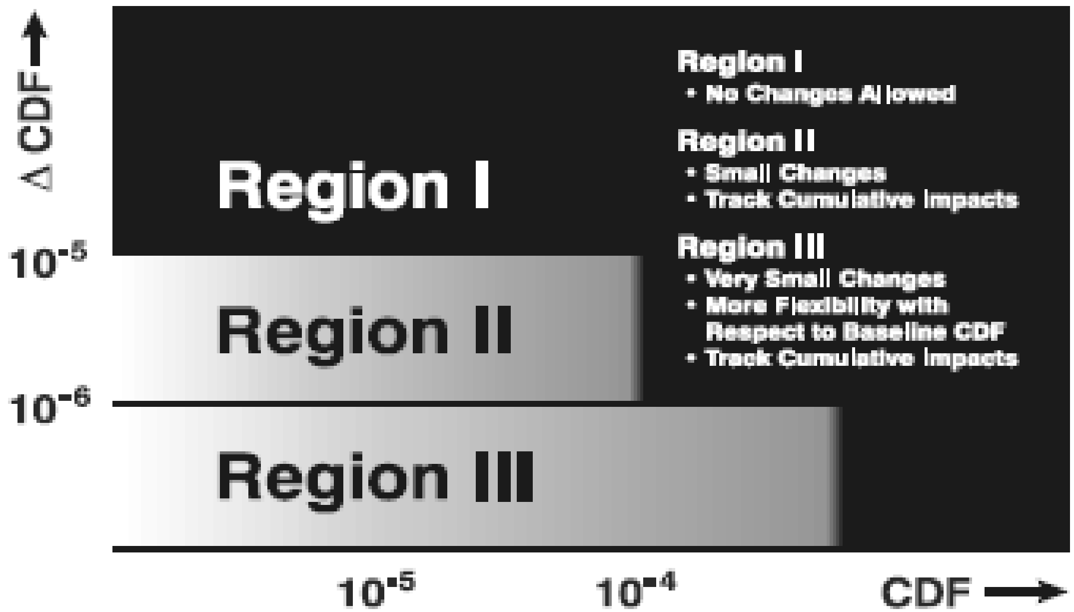
Acceptance Guidelines for Change in Core Damage Frequency



- A change in CDF of $1x10^{-6}$ (Region III) will be considered regardless of whether there is a calculation of total CDF, per Regulatory Guide 1.174.
- For this work, the calculated CDF is conservatively taken as a change from the baseline CDF.
- In addition, Large Early Release Frequency (LERF) would need to be evaluated, but was beyond the scope of this study.

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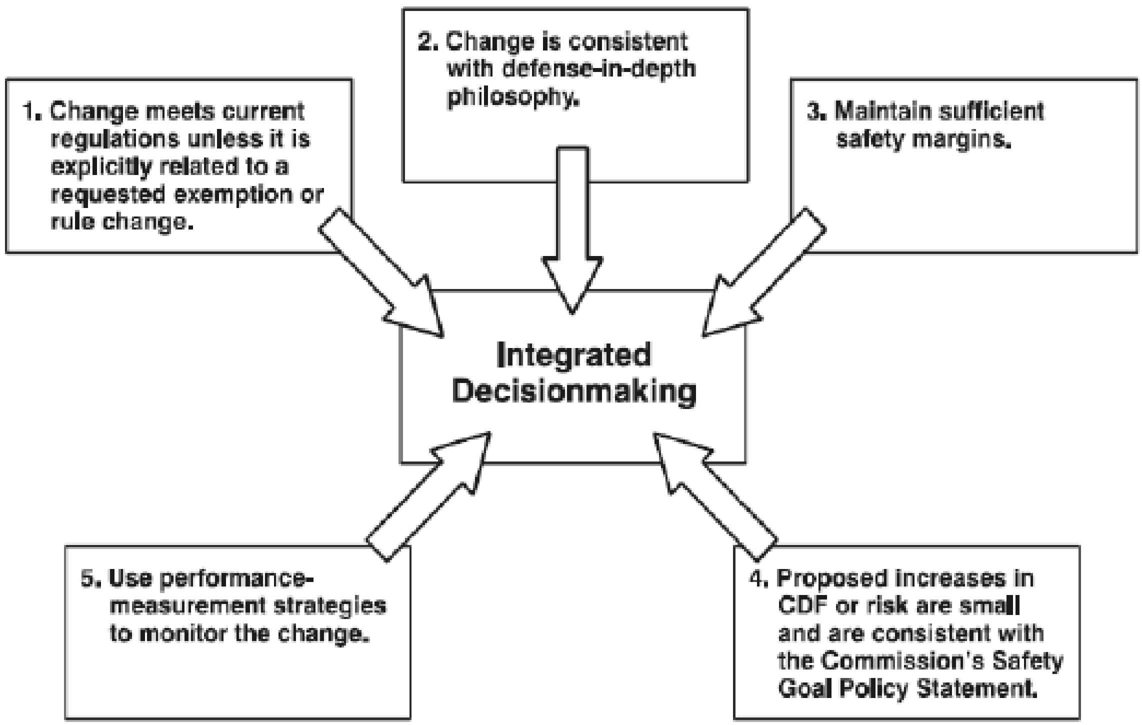
Suggested Approach for Development of a Revision to BTP 3-4

- A four phase methodology is suggested for postulation of HELB locations:
  1. A screening process would eliminate low consequence locations, consistent with a risk informed ISI (RI-ISI) approach.
  2. A systematic review of degradation mechanisms would be performed to identify those needing further evaluation.
  3. Mechanisms leading to pipe rupture where rapid propagation could occur are evaluated to establish whether effective mitigation strategies exist and will be implemented.
  4. For locations not able to be dispositioned in the prior phases, apply a NRC-approved method for probabilistic evaluation such as those applied for RI-ISI (or that described earlier). Risk insights from this evaluation would be used to reduce failure probability and/or mitigate consequences of failure, as appropriate.

Suggested Approach for Development of a Revision to BTP 3-4

The proposed methodology would be consistent with the NRC policy statement for the use of PRA methods and espoused principles outlined in Regulatory Guide 1.174 (see below).



Improved Break Location Postulation: Summary

- The current CUF criterion of 0.1 for postulated break locations has no clear technical basis.
- Continued use of this criterion could result in unnecessary costs without an associated safety benefit.
- Over 4 decades of industry experience have demonstrated that design transients do not result in high energy line breaks.
- Industry experience has been used to address uncertainties that existed when the current CUF criterion was established.
- 5 components were selected from NUREG/CR-6674 for evaluation, which provided a range of loads, material types and reactor designs.
- Use of leak probabilities (versus rupture) is conservative when considering postulated high energy line breaks.

Improved Break Location Postulation: Summary

- Initiation and leak probability calculations based on NUREG/CR-6909 were performed using pc-PRAISE.
- Core Damage Frequency is related to the leak frequency, consistent with the methodology used in NUREG/CR-6674.
- Resulting Core Damage Frequency (CDF) vs. CUF_{en} plots show no direct correlation between CUF_{en} and CDF values.
- Many current plants are designed to ANSI/ASME B31.1, which does not require calculation of CUF.
- For all of the selected components, a CUF_{en} of 1.0 resulted in a CDF $\leq 1 \times 10^{-6}$, which USNRC Regulatory Guide 1.174 considers very small and is well below the 1×10^{-4} value promulgated in NUREG-0800 Chapter 19.

Improved Break Location Postulation: Summary

- Based on the results of this study, if the use of CUF as a break location criterion is to be continued in combination with environmental fatigue analysis, a CUF_{en} of 1.0 can be used without an impact to safety.
- An approach that applies both deterministic and probabilistic elements is proposed.
- The proposed methodology is consistent with the NRC policy statement for use of PRA methods and the principles outlined in Regulatory Guide 1.174.
- Therefore, we recommend revising BTP 3-4 to apply this methodology.

Attachment 6

ENVIRONMENTALLY-ASSISTED FATIGUE SCREENING PRESENTATION

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Environmentally-Assisted Fatigue Screening

Process and Technical Basis for Identifying Environmentally-Assisted Fatigue Limiting Locations

(pending EPRI technical report)

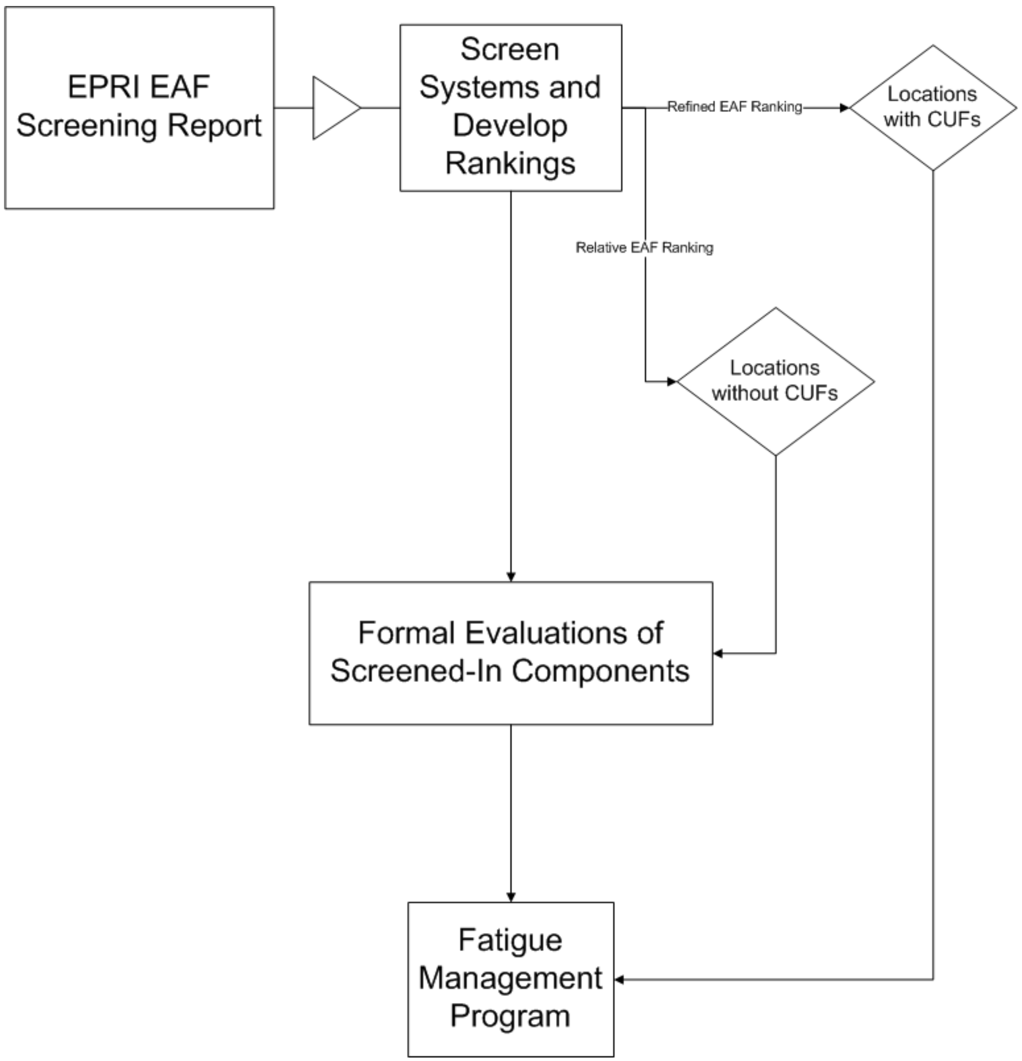
David A. Gerber, P.E.
Senior Associate

NRC Fatigue Meeting
January 5, 2012

Presentation Objective

- Present EPRI-sponsored proposed technical basis and process for environmentally-assisted fatigue screening.
- Show that approach described meets the need of GALL Rev. 2 for EAF screening.

Where Report Fits Into Overall Program



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General Objectives of EPRI Report

- Define a process for environmentally-assisted fatigue screening and ranking of components in nuclear power plant Class 1 systems.
- Describe the technical basis for the process
- This process:
  - must be effective for PWRs and BWRs, both with ASME Section III / B31.7 piping and B31.1 piping.
  - can be used to screen plant locations in order to rank them on the basis of environmentally-assisted fatigue. These ranked locations can then be compared to the NUREG/CR-6260 sample locations and may augment a plant's Fatigue Management Program (FMP).
- The desired outcome of this process is to determine plant locations which can be demonstrated to bound other locations of like materials and can serve as limiting environmentally-assisted fatigue locations for the plant.

Challenge

- License Renewal rules require that applicants demonstrate fatigue management including environmentally-assisted fatigue effects.
- The NUREG/CR-6260 locations were identified as a sample of locations to evaluate for environmentally-assisted fatigue and to include in the plant Fatigue Management Program.
- The challenge is to know if the NUREG/CR-6260 locations bound the plant for environmentally-assisted fatigue effects.
- If the NUREG/CR-6260 locations do not so bound, add bounding locations to FMP.

Criteria for Process

- The screening and ranking procedure developed has the following properties:
  - No requirement for new formal stress or fatigue analysis.
  - Includes procedures that are practical to use, with readily available design input.
  - Provides appropriate *relative* environmentally-assisted fatigue rankings of components.
  - Allows the use of either NUREG/CR-5704 (stainless steel)/6583 (carbon and low alloy steel)/6909 (Ni-Cr-Fe) or just NUREG/CR-6909 for all materials.

Benefits

License Renewal

- This process will enable plant owners to demonstrate knowledge of the locations in their plant that can serve as limiting locations for environmentally-assisted fatigue evaluations.
- This process provides the rationale for selecting these bounding locations.
- Plant owners will minimize the necessity of formal fatigue analysis, while meeting the regulatory requirements for determining the bounding environmentally-assisted fatigue locations in the plant.
- NRC staff can examine one possible uniform approach to determination of limiting locations for EAF evaluations for license renewal applications.

Approach

- Need relative measure for comparison.
- Not resorting to simplifications to identify locations with complex loading.
- Estimate of CUF and F_{en} necessary without requiring formal stress/fatigue analysis.
- Some complicating features are present.

More Challenges - CUFs are Not Equal

- Determination of this list of bounding locations is not as straight forward as multiplying each design CUF value by a factor or factors. Examples of the complicating factors are:
- Not all CUF values represent the same degree of analytical rigor.
  - Analysis of design severity plant transients produces different CUF values for a component than analysis of actual severity plant transients.
  - Analysis using “bundled transients”(1) yield significantly higher CUF values than analyses of the same component with “un-bundled” transients.
- For a given plant transient, F_{en} factors often will trend counter to the computed CUF values, thus potentially complicating the ranking of the CUF_{en} (CUF considering environmental influence on fatigue) values for a component.
  - Faster rise times for a thermal transient will tend to produce lower F_{en} factors, but larger CUF values. Since $U_{en} = F_{en} \times U$, the product of the two is not known a priori without further analysis.
- Analysis of design numbers of plant transients can yield different rankings of CUF and CUF_{en} values than analyses of projected numbers of plant transients.
  - The two different mixes of plant transients, each with their unique transient characteristics, can cause the weighted F_{en} factors and CUF_{en} values to vary significantly.

(1) *Bundled Transients*: Enveloping of multiple plant transients by one conservative plant transient.

CUF_{en}'s Vary by Material and DO

- Different materials of construction exhibit different environmentally-assisted fatigue characteristics, even in the same component.
  - The same plant transients applied to one component will produce different U_{en} values for different material of construction.
  - DO content affects materials of construction differently and varies by NUREG rule.

More Differences

- Further factors that influence the evaluations are:
  - Use of the alternate rules of NUREG/CR-5704 (stainless steel) and NUREG/CR-6583 (carbon and low alloy steel) will produce somewhat different values of F_{en} than the newer rules of NUREG/CR-6909 for those materials.
  - Components in similar plants will likely have similar estimated CUF_{en} characteristics, although some may have computed CUF values and others may not. This conclusion is based on an EPRI review of piping fatigue [1] where it was determined that:
    - Although ANSI B31.1 and ASME Code, Section III, Class 1 piping rules are fundamentally different, experience in operating plants has shown that piping systems designed to B31.1 are adequate.
    - The operation of B31.1 plants is also not different from that of plants designed to ASME Code, Section III, Class 1.

[1] EPRI Report "Fatigue Comparison of Piping Designed to ANSI B31.1 and ASME Section III, Class 1 Rules," TR-102901, EPRI, Palo Alto, CA, December 1993.

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Solution and New Terms

- Provide a robust solution without necessarily requiring a complete reanalysis. EPRI has developed a process for screening all the fatigue-sensitive components in a plant by ranking them in terms of CUF_{en} and then determining a set of *Sentinel Locations* such that each plant component is covered by one or more sentinel locations.
    - A *Sentinel Location* is a specific location in a piping system or vessel that serves as a leading indicator for environmentally-assisted fatigue damage accumulation. Sentinel locations are expected to accumulate more CUF_{en} than other locations and remain bounding as plant transients occur in plant life.
    - A *Thermal Zone* is defined as a collection of piping and/or vessel components which undergo essentially the same group of thermal and pressure transients during plant operations.

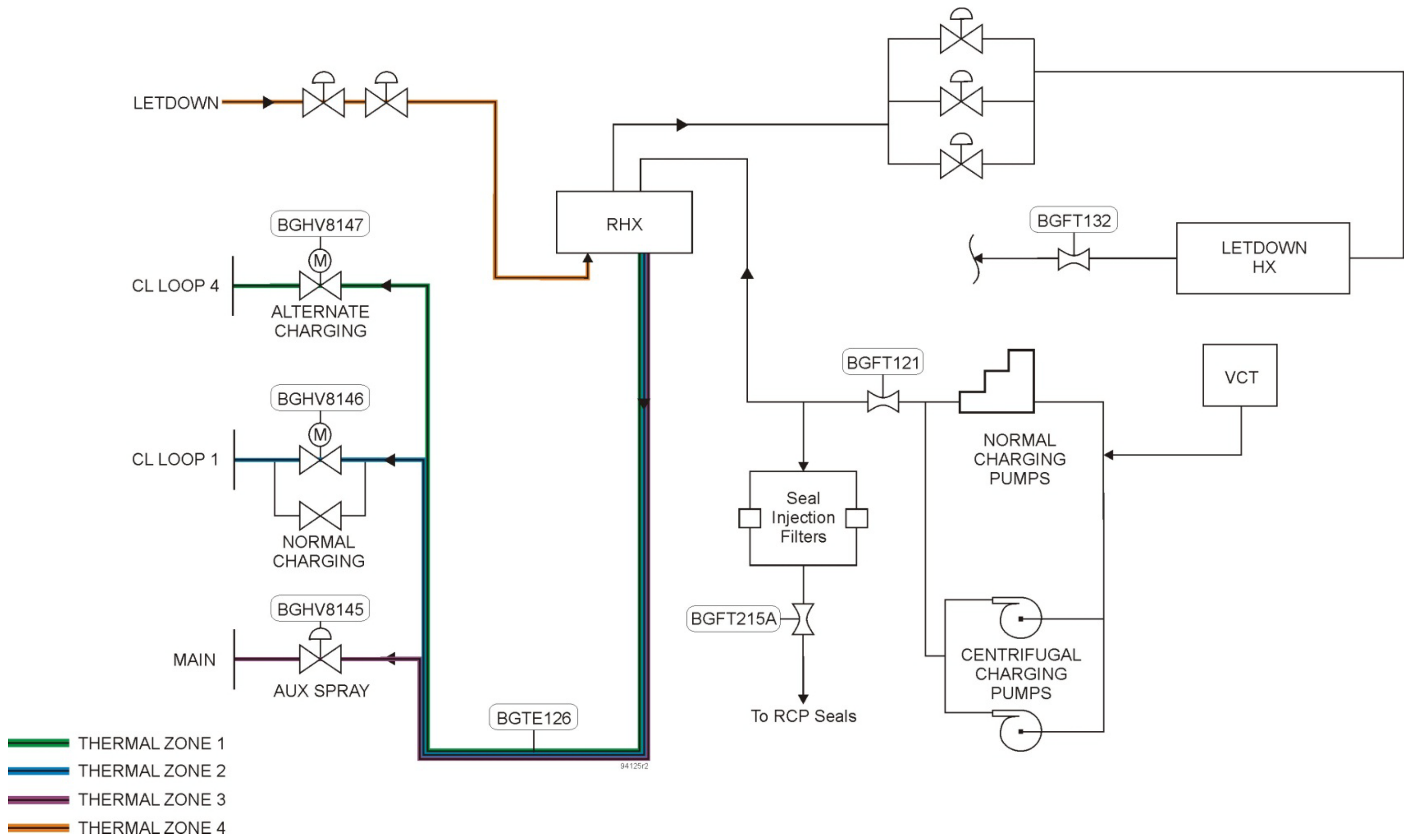
Sentinel Locations

- The sentinel locations in a thermal zone may be thought of as a Peloton, which refers to a densely packed group of bicycle racers. A leader is established, but over time a new leader may emerge as the transient mix accumulates.

Peloton

Thermal Zones

▪



Process Basis

- For the reasons discussed, it is necessary to evaluate components and/or locations in a component on a *uniform common basis* to accomplish valid ranking and identification of sentinel locations in each thermal zone. Plants with explicit fatigue design bases (have CUF values) can have:
  - Sets of components evaluated to a reduced, “bundled” set of plant transients and/or a mixture of bundled and unbundled transients.
  - Components or locations in components evaluated to additional refined analyses (e.g., elastic-plastic analysis) while other components or locations are not.
- To assure uniform determination of relative fatigue accumulation, these differences must be accounted for or eliminated. The screening processes are designed to make this common basis determination.

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Process Basis

- The screening process is used to review all Class 1 plant components susceptible to environmentally-assisted fatigue, categorize them into thermal groups, and identify one or more sentinel locations for each thermal group that can be analyzed and monitored for environmentally-assisted fatigue usage.
- The idea of sentinel location extends the basic approach that was used in NUREG/CR-6260 of analyzing a few challenging locations to represent the entire plant, but adds a semi-quantitative ranking system to demonstrate that each plant component is represented by at least one sentinel location.
- The idea of a thermal zone has been used for many years in piping analysis to both group and differentiate locations based on operating transient conditions.

Process Flow

- Make F_{en} estimation
  - Qualitative estimate of strain rate
  - Develop expected F_{en} (F_{en}*) as the average of the Best Estimate F_{en} for leading transient and the Maximum F_{en}
  - Compute U_{en}* for locations
- Compute estimated CUF (U*) and estimated CUF_{en}(U_{en}*)
  - Select leading transients
  - Compute thermal through-wall stresses
  - Extract bending moment and seismic stresses from DSR
  - Evaluate leading load pairs and determine estimated U* and U_{en}* Rank locations by U_{en}*, material type, thermal zone and compare to 6260 locations

Outline of Process Steps

This screening process consists of four stages:

1. Data Collection
   – Component geometry and material properties, plant transient characteristics and projections of plant transients for the licensed operating period.
2. Determination of Thermal Zones
   – Components are assigned to appropriate thermal zones and evaluated as a group. This allows definitive rankings to be determined.

Outline of Process Steps

3. Evaluation of Locations
   - Establish *relative* stress, CUF and CUF_{en} values.
   - Common basis approach.
   - Mitigates skewing effects of refined analyses (such as elastic-plastic analysis) for selected components.
   - Ranking on a common basis assures most highly stressed and cycled locations in each thermal zone are identified as leading indicators of fatigue damage for the thermal zone.
4. Ranking and Identification of Sentinel Locations
   - An estimated U_{en}* is determined.
   - Locations within each group with the highest estimated U_{en}* are reviewed to determine one or more sentinel locations.

Process Development Assumptions and Characteristics

- Thermal zones are employed to provide consistency in development of estimated F_{en} values and common basis stress approximations.
- Common analytical basis (un-bundled transients) is used to put all analyses in a thermal zone on the same transient basis.
- Calculated plant piping loads and stresses are used instead of piping attachment point umbrella loads.
- Design severity transients (can use actual severity, if available and consistently applied) are used.
- Geometric factors are applied to stress terms.
- Materials of construction are evaluated together as a group in each thermal zone.

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Process Development Assumptions and Characteristics

- Several assumptions are inherent in the process:
  - An estimated F_{en} method is sufficient for a screening process; this process is not intended to provide an ASME qualification of components.
  - Several characteristics of the process are important.
    - Linear elastic stress analysis and superposition of stress contributions are used.
    - The F_{en} factor is applied only for increasingly tensile portions of transients, based on the guidance of MRP-47 [1].
    - The K_e factor is included in both the determination of strain range and estimated strain rate (consistent with proposed ASME-Code Case N-792-1).

[1] Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application, MRP-47, Revision 1, September 2005.

Result of Screening Process

- The result of this screening process is a listing of fatigue-sensitive reactor coolant pressure boundary components, organized into groups, ranked by CUF_{en} severity, with at least one sentinel location identified for each group of components.

Evaluation of Locations

- Two analytical evaluation procedures are developed to aid in the evaluation process:
  - one to perform *F_{en} Estimation Evaluations*
    - For plants with explicit fatigue design basis (CUFs) (e.g., Section III or B31.7 piping)
  - one to perform *Common Basis Stress Evaluations*
    - For plants with explicit fatigue design basis but non-uniform transient bundling
    - For plants without explicit fatigue design basis (e.g., B31.1 piping)

F_{en} Estimation Evaluation Procedure

F_{en} Estimation Evaluation Procedure

- Used to estimate F_{en} for locations in plant components on the basis of the relevant parameters – dissolved oxygen, maximum temperature and estimated tensile strain rate – of the leading transient(s).
- The procedure is developed to use:
  - For plants with and without explicit fatigue design analyses available.
  - With design transients or actual transients (consistently applied).
  - With design numbers of transients or licensed operating period (e.g., 60-year) projected numbers of transients.

Technical Basis for F_{en} Estimation Evaluation

- For screening, the rules for calculating F_{en} values may either be taken from NUREG/CR-5704 for stainless steel material, NUREG/CR-6583 for carbon/low alloy steel material and NUREG/CR-6909 for Ni-Cr-Fe material, or from NUREG/CR-6909 for all materials.
- These rules allow calculation of F_{en} factors based on the material at the postulated failure location (SS, CS, LAS and Ni-Cr-Fe) and the following environmental parameters:
  - Estimated strain rate ($\dot{\varepsilon}$), during the transients, in [%/sec].
  - Concentration of dissolved oxygen (DO) in the water, in [ppm].
  - Maximum fluid/metal temperature (T) during the transients, in [°C].
  - (Note: sulfur content of the metal (S) is also a factor for CS and LAS. However, this procedure will conservatively assume all CS/LAS components have the worst possible sulfur content.)

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Technical Basis for F_{en} Estimation Evaluation

- Since the procedure is an aid to a screening evaluation for relative ranking, exact values of these parameters are not calculated from qualified design input. Instead, estimated values are determined based on familiarity with operation of the various plant systems and components during both normal operation and the transient conditions as defined in the plant design specifications. Specifically:
  - Any components which have no exposure to the “environment” (i.e., heated primary coolant water) are assigned an F_{en} value of 1.0.
  - Any transients associated with fast transients (e.g., seismic) may be assigned an F_{en} value of 1.0.
  - A qualitative estimate of the strain rate ($\dot{\varepsilon}$) for the controlling fatigue transient(s) will be determined, based on knowledge of the corresponding plant system. Each component will be identified with one of eight possible categories shown in Table 3-1.

Technical Basis for F_{en} Estimation Evaluation

Table 3-1
Strain Rate Categories

	Strain Rate Category	Estimated $\dot{\varepsilon}$ [%/sec]
	Extreme	≥ 5.0
	V.High	~ 1.3
	High	~ 0.33
	Mid-High	~ 0.087
	Medium	~ 0.023
	Low-Mid	~ 0.0059
	Slow	~ 0.0015
	V.Slow	≤ 0.0004

Technical Basis for F_{en} Estimation Evaluation

- An estimated DO value of “Low” (≤ 0.04 ppm) will be applied for all components exposed to reactor water for PWRs. This determination is based on the observation that for the entire history of most PWRs, the concentration of dissolved oxygen is maintained below 0.04 ppm at all times when water temperature is ≥ 150°C (302°F) (with rare exceptions). (Note: when water temperature is below 150°C, DO is no longer a factor in the value of F_{en} for any of the materials considered in this procedure.) For BWRs, the DO values must be determined based on the procedural policies of the plant for water chemistry control.
- An estimated upper-bound T value will be determined based on the collected design transients for the respective plant systems (for NUREG/CR-6909 evaluations, an average T value is used).

Technical Basis for F_{en} Estimation Evaluation

- For each component, this evaluation computes two hypothetical F_{en} values, one using the estimated parameter values described above, and the second using the same estimated values for DO and T, but using the worst possible (i.e. most conservative) value for strain rate.
- These two computed values are averaged to produce an expected F_{en} for each component. This two-part expected F_{en} is based on experience with performing detailed F_{en} analyses; in general, the estimated F_{en} from a detailed analysis is close to the F_{en} value computed for just the controlling transient pairs, but slightly higher due to contributions from the less-significant fatigue pairs.
- A simple average is judged to magnify the contributions of the less-significant transient pairs to yield a reasonably conservative value suitable for ranking without performing a detailed analysis.

Common Basis Stress Evaluation Procedure

Common Basis Stress Evaluation Procedure

– Procedure based on the rules of ASME NB-3600 modified to address a screening evaluation for relative ranking of locations. Rationales for this approach are that:
  - Majority of the components in the screening population are piping components for which the rules of NB-3600 are appropriate.
  - NB-3600 equations are explicitly defined and require minimal analyst interpretation so that they can be easily included in a spreadsheet.
  - NB-3600 rules are representative of the more general rules of ASME NB-3200 design by analysis, which are appropriate for Class 1 plant components.

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Common Basis Stress Evaluation Procedure

- Used for components where:
  - no explicit design fatigue analysis is available, or
  - where is it desired to put components with a fatigue analysis on a common basis
- The user will estimate a common basis CUF.
- The Common Basis Stress Evaluation Procedure is used to perform the following stress computations to determine the common basis CUF:
  - Through-wall transient thermal stresses are computed for leading transients. Transients with thermal shocks are found to be the leading fatigue usage contributor in piping and vessel stress analyses.
  - Piping moment range stresses and pressure stresses are extracted from the plant piping Class 1 stress report. Use of actual piping results avoids the use of piping umbrella loads and helps differentiate moment loadings for locations within a piping system.
  - Peak stresses at discontinuities are accounted for using SCF/FSRFs taken from the ASME Code.

Rationale for the Common Basis Stress Evaluation Procedure

- Taking guidance from the EPRI Fatigue Management Handbook [1], formulas have been developed to compute stresses arising from maximum transient through-wall temperature distributions, axial temperature differences, thermal and mechanical bending stresses and geometric characteristics for piping and vessel components. These formulas ensure a common level of analysis so that the computed stresses are directly comparable between locations.
- These formulas assume that the stresses are linear elastic, and so may be combined using linear superposition. Non-linear plasticity effects are accounted for using elastic-plastic penalty factors (K_e) in accordance with ASME Code Subarticles NB-3200 and NB-3600. Use of linear elastic rules for computing CUF retains technical parity among the components in a thermal zone. By contrast, using elastic-plastic non-linear techniques in a fatigue analysis may significantly reduce the computed CUF for that component, which would give it a much lower CUF than other locations with comparable fatigue duty.

[1] Materials Reliability Program: Fatigue Management Handbook, MRP-235, Revision 1 (with corrections)

Rationale for the Common Basis Stress Evaluation Procedure

- The linear elastic stress state for a location may be computed as the linear summation of the individual stresses caused by various types of loads. Most pressure vessels and piping system components include stresses due to internal pressure, thermal (due to temperature distribution in the component), and boundary interface loads, such as forces and moments caused by thermal expansion, thermal stratification, anchor displacement, seismic movement, etc. Deadweight and residual stresses may be ignored, because they do not vary with time and therefore do not impact the computed stress range.
- For a linear elastic stress analysis, stress contributions may be classified as one of two types:
  - Stresses due to loads, such as pressure, piping thermal expansion, etc. that are directly scalable to pertinent parameters (pressure, temperature, etc.), and
  - Time-dependent thermal stresses, which depend on the axial and radial temperature distributions in the component rather than any single instantaneous parameter.

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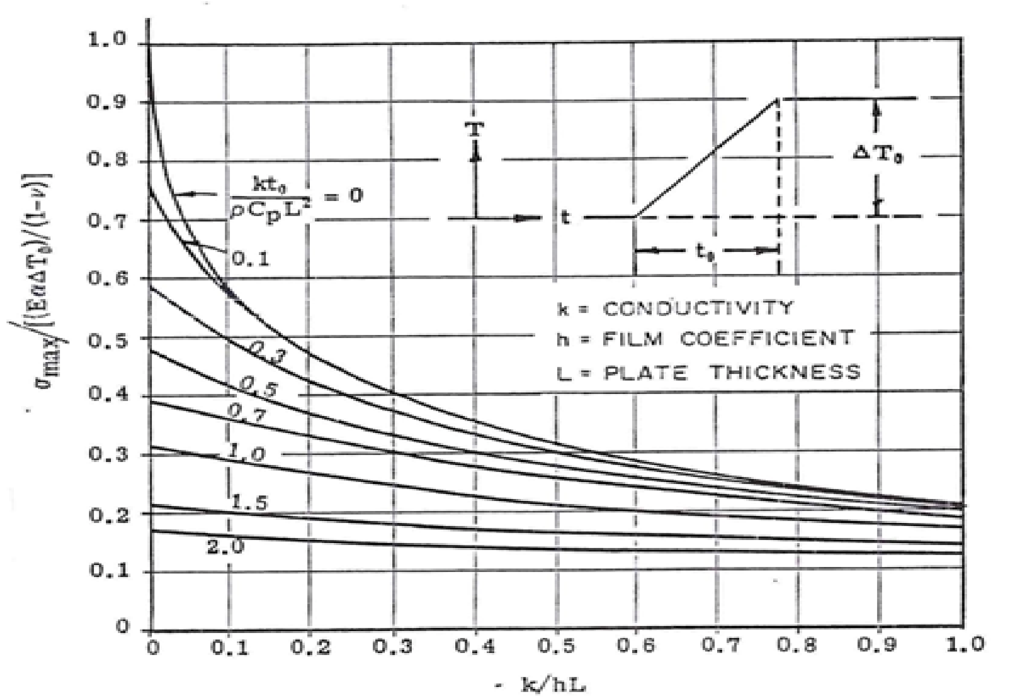
Rationale for the Common Basis Stress Evaluation Procedure

- Stress contributions of the second type depend on the temperature history and are typically calculated by a time integration of the product of a predetermined Green's function, or influence function, and the transient temperature data. Performing this integration is more complex than is desired for this screening process. Instead, an estimate of the maximum stress range during each significant thermal transient is computed, as described below. This estimate applies a uniform level of conservatism, and is sufficiently precise to determine a relative ranking among the components in a thermal zone.
- The stress computation combines stresses from the following terms:
  - Through-wall transient thermal stresses are computed using the graph shown in Figure 3-1. For each transient, two non-dimensional factors ($k/(hL)$ and $(kt_0)/(\rho c_p L^2)$) are computed as entry into the curve for the determination of the normalized thermal peak stress.
  - Piping moment range and pressure stresses are extracted from the plant piping Class 1 stress report. Umbrella loads (conservative loads assigned to the system to facilitate design of adjoining systems) are not recommended, as they don't inform the relative severity at different locations.
  - Thermal stratification moment stresses are assumed to be negligible or included in the computed piping moment stress range.
  - Seismic stresses.
  - Peak Stresses at discontinuities are accounted for using appropriate SCFs.

Rationale for the Common Basis Stress Evaluation Procedure



T = temperature
ρ = density
C_p = specific heat
L = pipe wall thickness

(Note that all parameters shown must remain dimensionless)

Figure 3-1 Determination of Transient Stresses for Ramp Transients

Rationale for the Common Basis Stress Evaluation Procedure

- The *Common Basis Stress Evaluation Procedure* is used to determine approximate stress ranges arising from pairs of selected significant transients, compute alternating stress values including simplified elastic-plastic (K_e) effects, and produce incremental CUF (U_{incr}) for input numbers of plant transients (either design numbers or projected numbers).
- These incremental CUF values are added to produce the common basis CUF (U*). Estimated F_{en} values are computed (using either the older or newer environmentally-assisted fatigue rules), along with an incremental U_{en} for each transient pair.
- These are summed over the significant transients to yield an estimated U_{en}* for that location.

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Limitations and Assumptions of the Process

- Stresses caused by complex loading, such as thermal stratification, are not used in the Common Basis Stress Evaluation process. It is typically not practical to compute stratification stresses using this methodology. However, for components subjected to this type of loading, fatigue calculations are expected to have been performed already. Such is the case, for example, with PWR surge lines.
- Likewise, axial thermal gradient stresses produced by geometry or material transitions are also not considered in this process. Branch nozzles without thermal sleeves are commonly subject to stresses caused by axial thermal gradients. Such loading may be attributed to the injection of colder fluid into a hot header, giving rise to significant thermal stresses of a steady state nature near the nozzle corner. Sophisticated fatigue analyses are typically employed to disposition these types of components, and many of them, such as the charging and safety injection nozzles, are the NUREG/CR-6260 locations (the GALL report requires evaluation of the NUREG/CR-6260 locations at a minimum).
- The Common Basis Stress Evaluation process is valid for cylindrical or flat plate components being based on NB-3600 concepts.
- The process does not produce new formal stress results, but uses those results that are available in plant design reports.

Screening Process

1. Gather Required Inputs for Class 1 Vessels and Piping Systems.
2. Determine Thermal Zones for Each System.
3. Identify Materials and Candidate Locations.
4. Calculate U_{en}* Rankings for Each Candidate Location.
   A. F_{en} Estimation Evaluation Procedure
   B. Common Basis Stress Evaluation Procedure
5. For Each Material in Each Thermal Zone perform ranking and sentinel location identification.
6. Evaluate Next Candidate Location.
7. Evaluate Next Thermal Zone.
8. Evaluate Next System.
9. Compile Final List of Sentinel Locations.

Conclusions

- Report provides technical basis of the screening process used to evaluate a plant to determine EAF limiting locations for fatigue monitoring. Procedures for this screening evaluation are described and applied to a pilot PWR plant.
- Process designed to equip license renewal applicants with a consistent method to identify EAF limiting locations additional to the sample locations evaluated in NUREG/CR-6260 for their reactor type and vintage.
- Guiding principles for the screening and ranking process included:
  - Consistent technical basis.
  - Analytical method using readily available design input from P&IDs, piping isometric drawings and piping stress reports.
  - Only basic stress or fatigue analysis required.
- The following are the basic areas of new technology developed by this project:
  - Procedure for Estimating F_{en} Factors.
  - Procedure for Estimating U_{en}.
- An example of the process is provided.

Questions and Comments

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Attachment 7

THERMAL FATIGUE MANAGEMENT GUIDELINE FOR NORMALLY STAGNANT NON-ISOLABLE RCS BRANCH LINES PRESENTATION

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Thermal Fatigue Management Guideline for Normally Stagnant Non-Isolable RCS Branch Lines

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January 5, 2012

Introduction

- The EPRI MRP has an ongoing program (MRP-146) to assist PWR owners manage thermal fatigue concerns in normally stagnant, non-isolable reactor coolant system branch lines
- MRP-146 contains “Needed” requirements as part of the NEI 03-08 materials initiative
- The EPRI MRP has met with NRC previously to present these industry efforts:
  - May 2005 (MRP-146 program discussed)
  - April 2009 (MRP-146S supplement discussed)

Introduction (2)

- Since the 2009 meeting:
  - PWR fleet surveyed and plant screening information obtained
  - Initial inspections for all screened in branch lines have been completed
  - MRP-146R1 was published (June 2011)

Introduction (3)

- Purpose of this meeting:
  - Inform NRC of important developments since last meeting on this subject
  - Provide overview of MRP-146R1 and implementation plan
  - Summarize ongoing activities
- Entertain comments and discussion
  - NRC approval is not being requested

Presentation Content

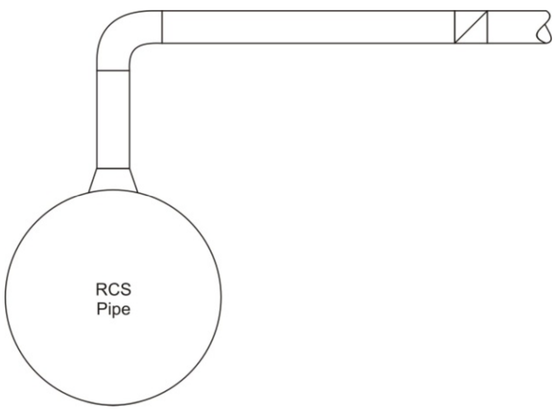
- MRP-146 program background
- MRP-146S implementation experience
- Overview of MRP-146R1 changes
- MRP-146R1 “Needed” requirements in revised branch line assessment methods
- Summary of improved MRP-146R1 guidance for:
  - Monitoring
  - Inspection
  - Mitigation
- MRP-146R1 implementation and next steps
- Conclusions

MRP-146 Program Background

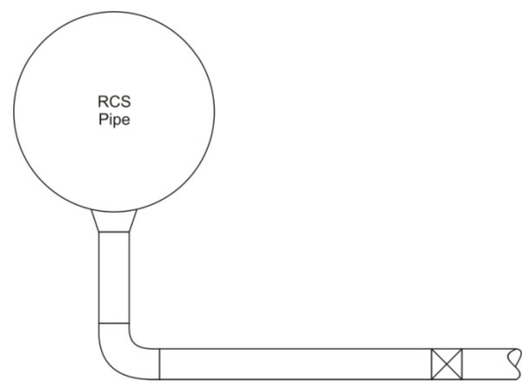
MRP-146 Program Background
Line Configurations

Branch lines are categorized into three basic configurations depending on attachment to RCS piping:

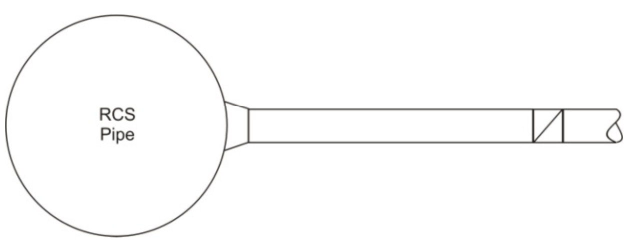
Up-horizontal (UH) configuration



Down-horizontal (DH) configuration



Horizontal (H) configuration



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MRP-146 Program Background

EPRI Thermal Fatigue Project History

- EPRI Thermal Stratification, Cycling and Striping (TASCS) Program began in 1989
  - Response to NRC Bulletin 88-08
  - Final report issued in March 1994
- Industry (NRC and Utility) concerns (1998)
  - Leakage events still occurring
  - TASCS methodology did not predict failure location of Farley event
  - Swirl penetration and stratification effects not well defined

MRP-146 Program Background
EPRI Thermal Fatigue Project History (2)

- EPRI/MRP formed the Thermal Fatigue Issue Task Group (ITG) in 1999 - established to proactively address concerns with pipe leaks in non-isolable piping attached to the RCS
- Interim guidance issued in 2001 (MRP-24)
  - Focus on lines which had exhibited leakage in service
  - Provided screening criteria based on experience and limited experimental work, inspection recommendations
  - Inspection interval & other potentially susceptible lines not addressed

MRP-146 Program Background

EPRI Thermal Fatigue Project History (3)

- Additional research continued on model development
  - Small scale phenomena testing
  - Review of available plant data and OE
- Model completed in 2004 (MRP-132)
- Management guideline published in 2005 (MRP-146)
- MRP fatigue efforts moved under the Technical Support Committee (TSC) in 2006
- Plant assessments completed in 2007 using EPRI QA software implementing the MRP-132 model (MRP-170)
- MRP-146 supplemental guidance published in January 2009 (MRP-146S)
- MRP-146R1 published in June 2011

MRP-146 Program Background
MRP-132 Analytical Model

- Provides general methodology for assessing branch line susceptibility to swirl penetration thermal cycling
- Analytical model based on:
  - Scaled model testing
  - Past leakage events
  - Plant monitoring data
  - Results from previous TASCS program
- Model addresses technical concerns regarding the TASCS program – much improved technical basis

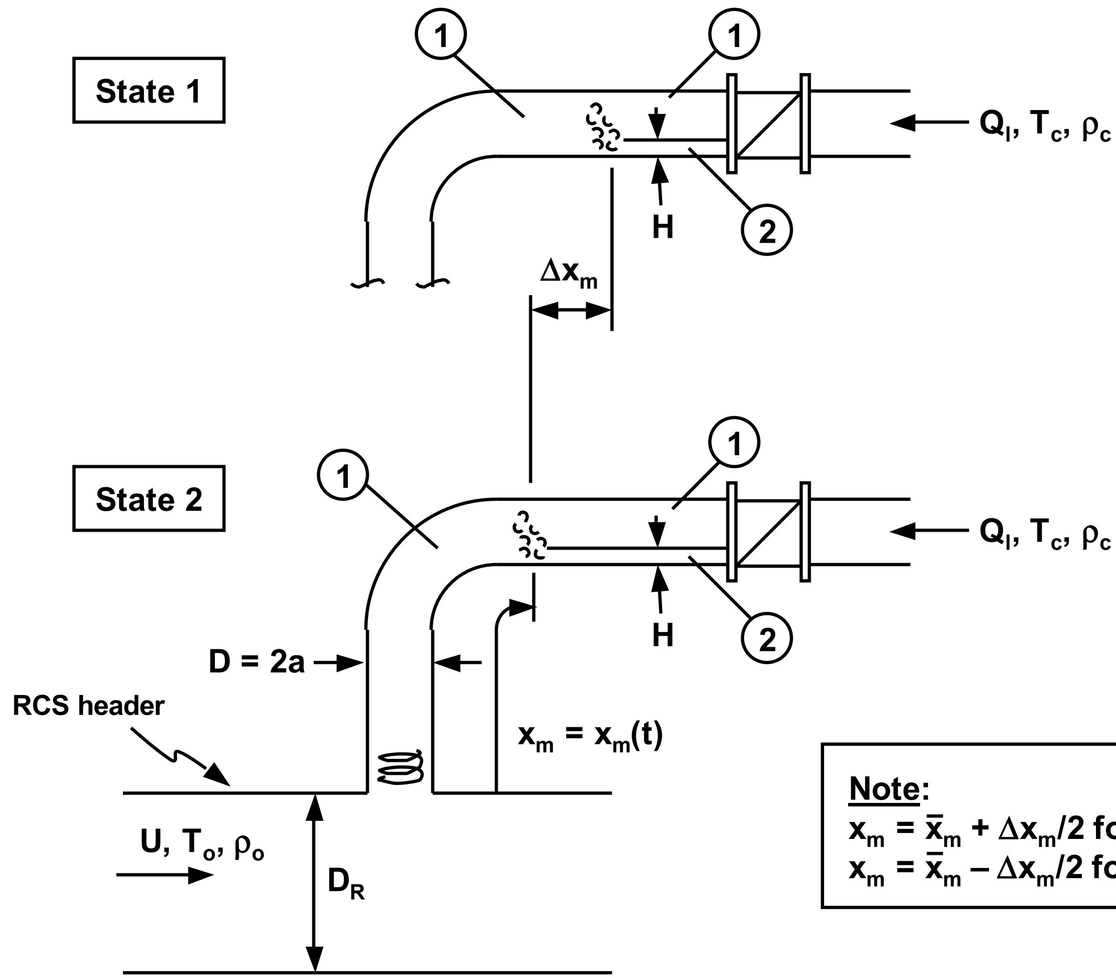
MRP-146 Program Background

MRP-132 Analytical Model (2)

- General method for thermal cycling assessment was developed
  - Screening: Is thermal cycling predicted to occur?
  - Evaluation: What are the thermal loads for structural analysis to determine inspection frequency?
- Branch line screening remains valid for UH/H/DH configurations
- For MRP-146R1, the thermal loading definition from MRP-132 is only in used in part for DH configurations

MRP-146 Program Background

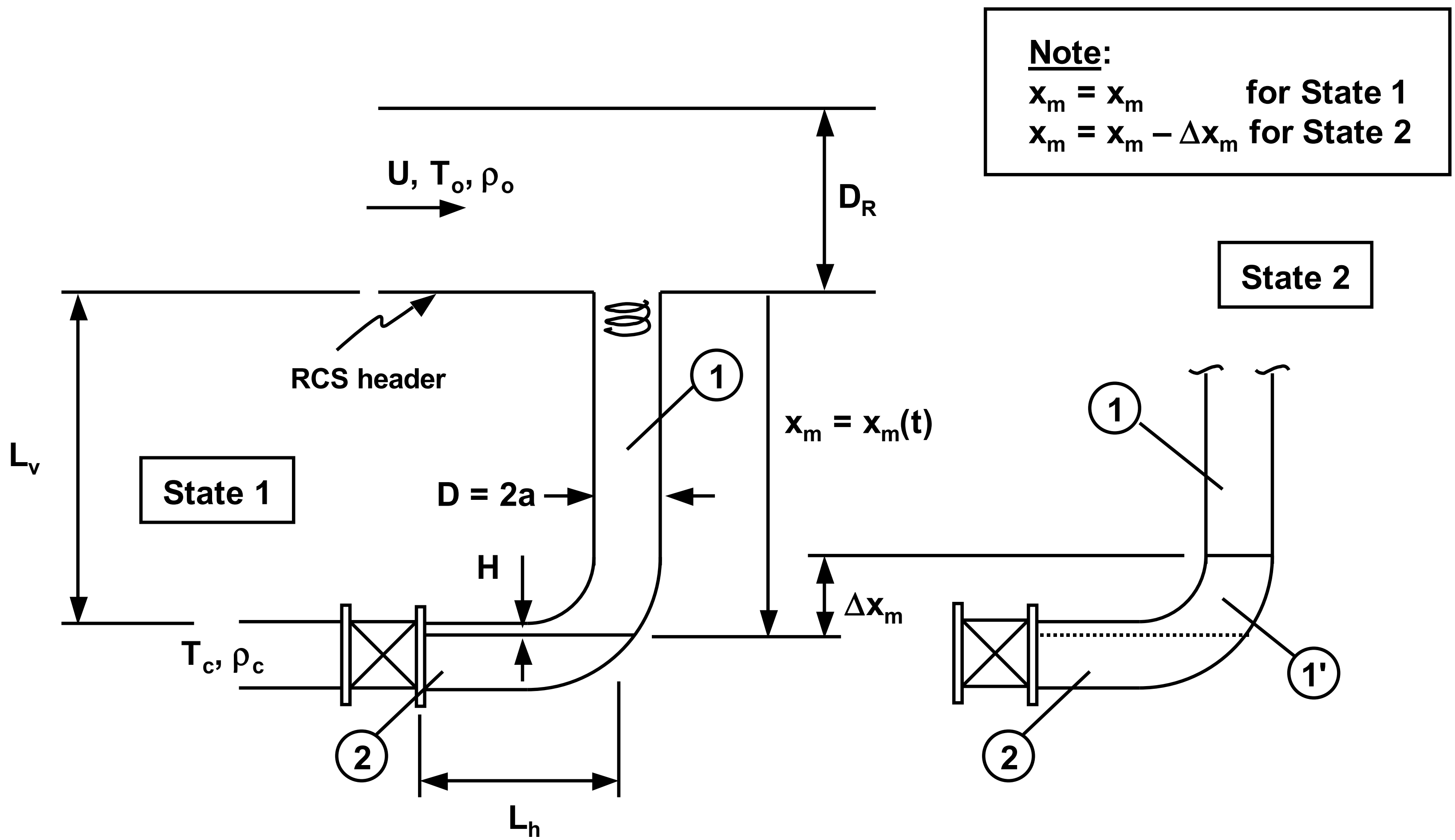
Review of Model for UH/H Lines



Note:
$x_m = \bar{x}_m + \Delta x_m/2$ for State 1
$x_m = \bar{x}_m - \Delta x_m/2$ for State 2

MRP-146 Program Background

Review of Model for DH Lines



MRP-146 Program Background
Model Branch Line Pre-screening

- Branch line pre-screening:
  - Branch line must be stagnant during normal plant operation
  - UH/H configurations must have in-leakage potential to screen in
  - Only UH lines greater than 2-inch NPS in scope
  - Only H and DH lines greater than 1-inch NPS in scope
- Screening criteria used to determine what lines require further consideration based on:
  - RCS and branch geometry
  - Operating conditions

MRP-146S Implementation Experience

MRP-146S Implementation Experience
Initial Inspections

- MRP-146S required that initial inspections be completed during the first refuel outage after January 31, 2009
- Initial inspections have been completed – cracking discovered in one drain line (thermal fatigue not confirmed as exclusive cause)
- No other indications reported
- Several follow-up inspections have already taken place – no indications reported

MRP-146S Implementation Experience

Drivers for MRP-146R1

- Utility implementation of MRP-146S uncovered several issues and areas needing further clarification:
  - Conservative fatigue analysis due to the high uncertainty associated with UH/H thermal loading resulted in high usage
  - For socket welded DH lines, conservative fatigue analysis due to the unknown quality of the elbow-to-horizontal pipe fillet weld resulted in high usage
  - Monitoring durations and data acquisition periods not prescribed
  - Inspection volume detail not adequate in some instances
  - Mitigation guidance limited

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Overview of MRP-146R1 Changes

Overview of MRP-146R1 Changes

- MRP-146R1 fully replaces MRP-146R0, but only supersedes some aspects of MRP-146S
- While there are significant changes from the previous requirements specified in MRP-146R0 and MRP-146S, there are no additional plant activities required by this revision that were not part of the MRP-146S requirements
- The timeframe for completing these requirements is also consistent with MRP-146S
- The changes reflect inquiries and lessons learned during the implementation of MRP-146S

Overview of MRP-146R1 Changes (2)

- MRP-146R1 changes include:
  - *Revised "Needed" requirements replacing Table 1-1 of MRP-146S*
    - Separate tables for UH/H and DH configurations
    - Date driven implementation schedule
    - More clearly written
  - *New UH/H assessment method*
    - Greater focus on in-leakage determination
    - Analysis no longer relied upon for establishing an inspection frequency
    - More clearly written

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Overview of MRP-146R1 Changes (3)

- *Revised DH assessment method*
  - Inspection frequency may be defined without analysis
  - Inspection frequency with analysis is reduced for some instances
- *Revised monitoring guidance*
  - Guidance more specific
  - Provides more options for valve leakage determination

Overview of MRP-146R1 Changes (4)

- *Revised inspection guidance*
  - Inspection volumes slightly modified
  - Examination volume details provided for socket welded branches and UH/H horizontal sections
- *Expanded thermal fatigue mitigation guidance*
  - Several plant modifications discussed w/ examples
  - Valve maintenance actions provided to help prevent leakage

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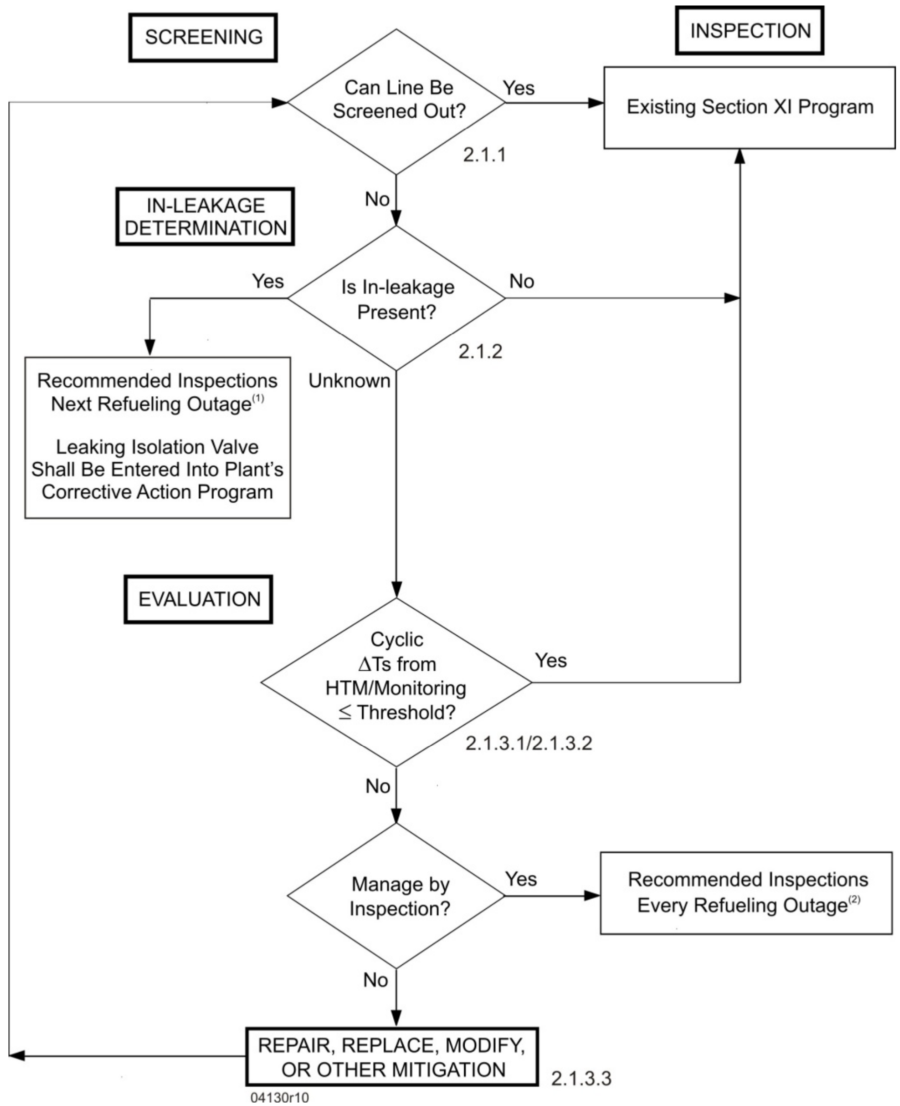
MRP-146R1 “Needed” Requirements in Revised UH/H and DH Assessment Methods

UH/H Assessment Method



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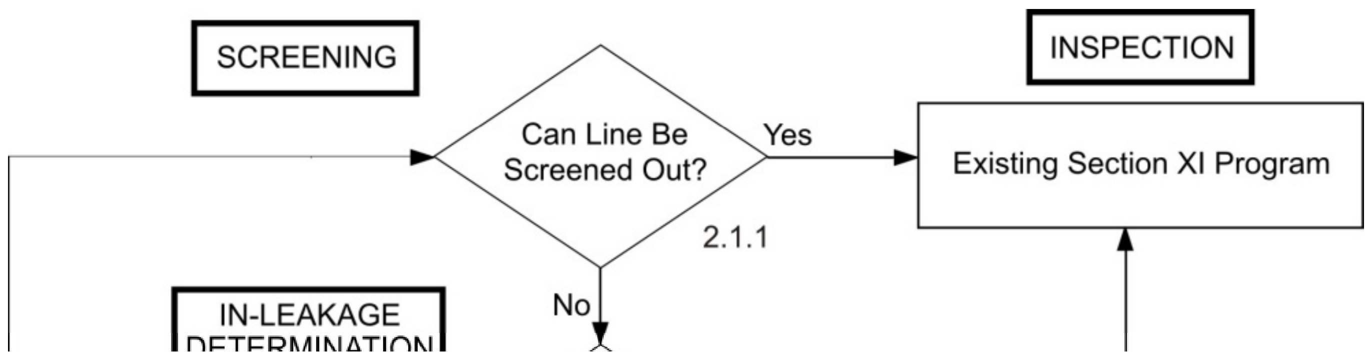
UH/H Assessment Method (2)

- Screening
  - Screening step remains unchanged from MRP-146R0 and MRP-146S



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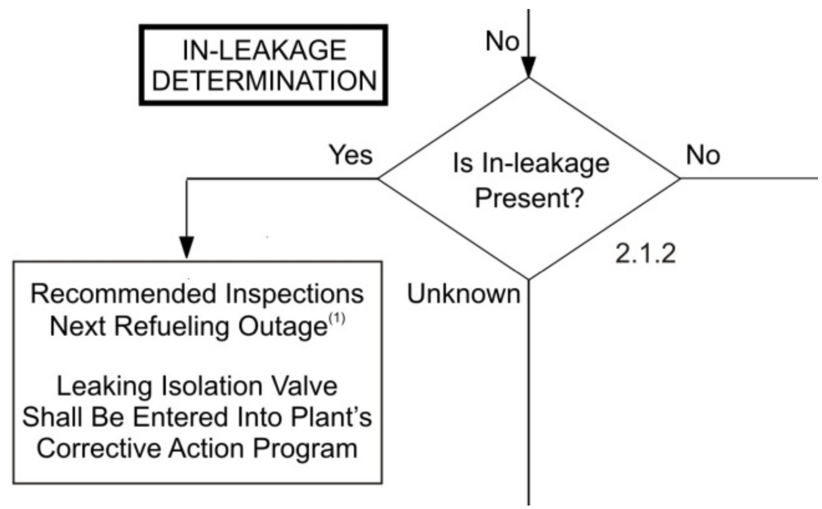
UH/H Assessment Method (3)

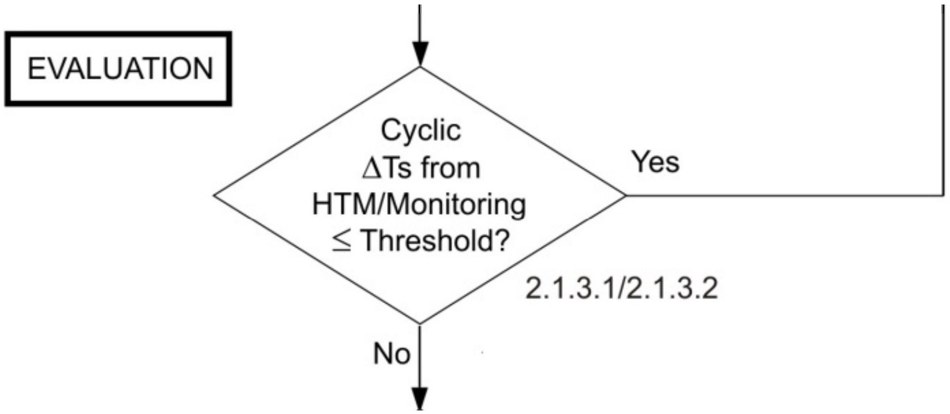
- <u>In-leakage Determination</u>
  - Primary element of management strategy
  - Two general methods:
    - Temperature monitoring
    - Valve leakage testing



It is understood that in-leakage may be found during an RFO or during startup such that a plant may operate for a full cycle before inspection is practical. While not specifically required, it is expected that efforts will be made to resolve the in-leakage issue prior to full power operation.

UH/H Assessment Method (4)

- <u>Evaluation</u>
  - Thermal cycling not significant if $\Delta T < \Delta T_{threshold}$ (ΔT from temperature monitoring or heat transfer modeling)
  - Unchanged from MRP-146R0 and MRP-146S



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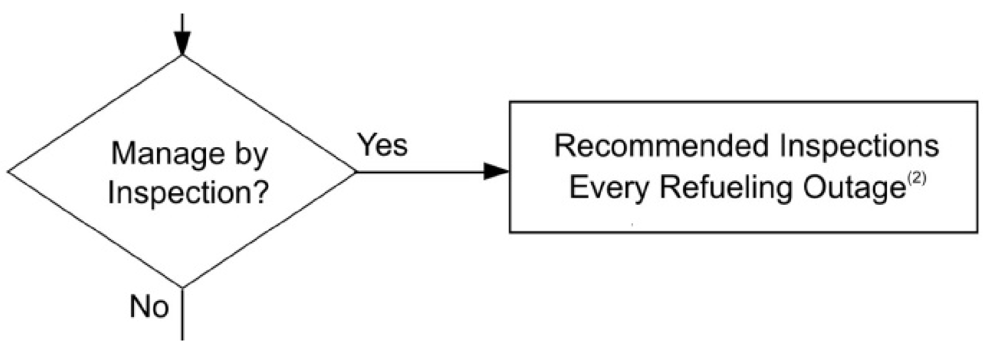
UH/H Assessment Method (5)

- Inspection
  - For significant thermal cycling, management by inspection (every refueling outage) is required
  - Analysis no longer relied upon for establishing an inspection frequency
  - High uncertainty associated with the thermal loading the driver for now specifying an inspection frequency



UH/H Assessment Method (6)

- Alternate Actions
  - Actions to mitigate thermal cycling loadings remains an option
  - Section 2.5 of MRP-146R1 significantly expanded to address thermal fatigue mitigation

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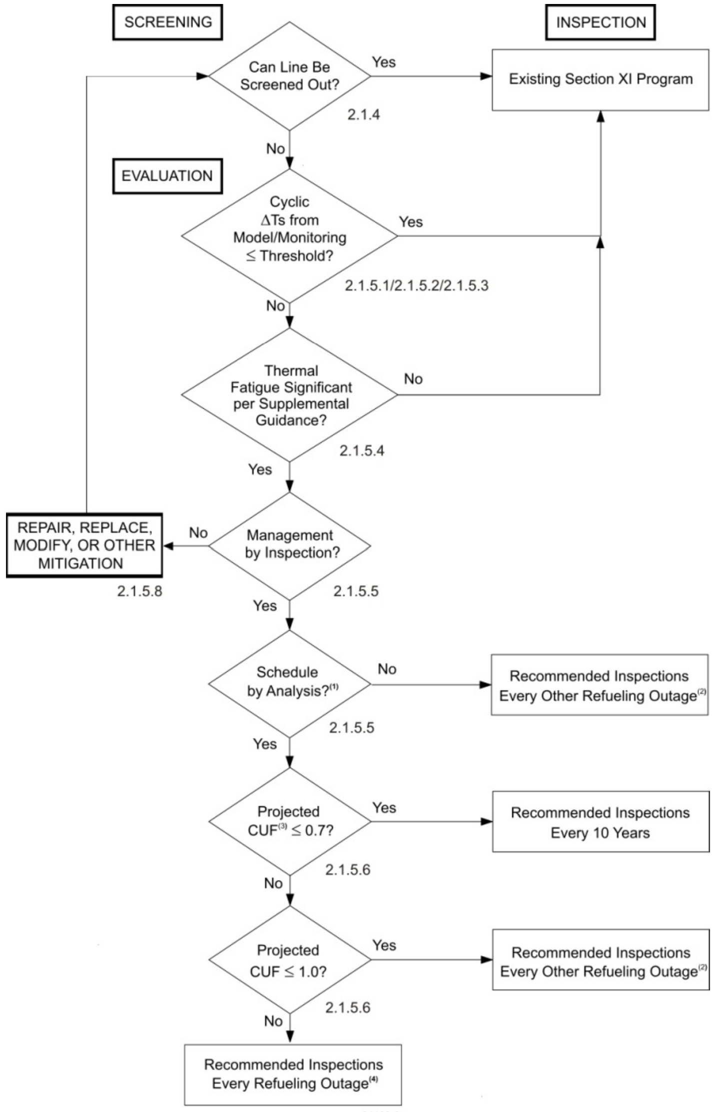
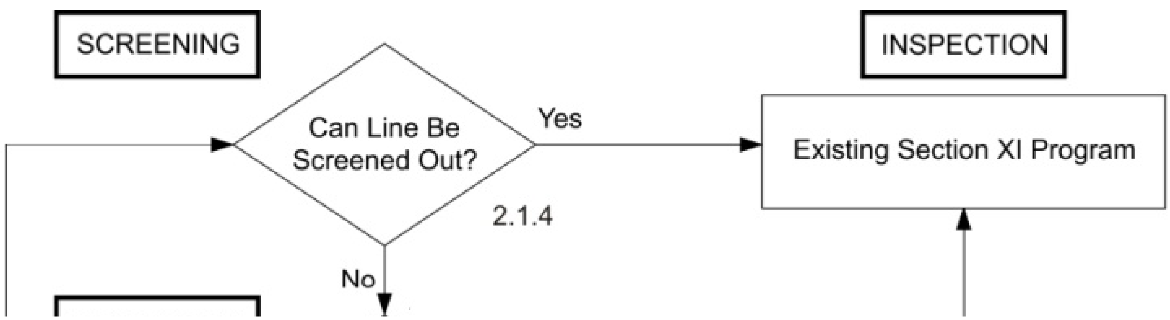
DH Assessment Method



DH Assessment Method (2)

- Screening
  - Screening step remains unchanged from MRP-146R0 and MRP-146S



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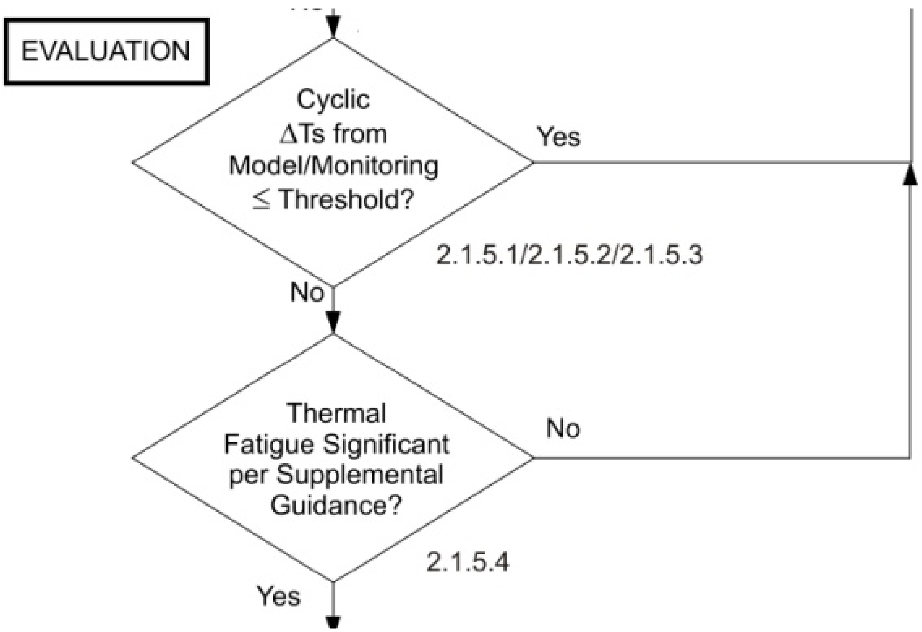
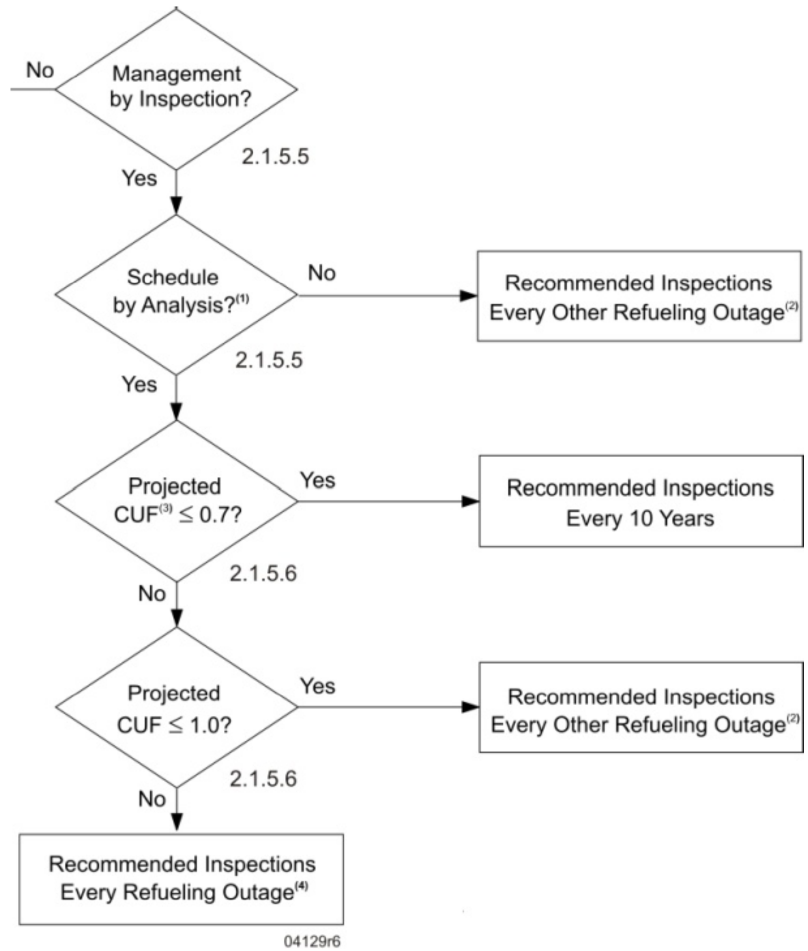
DH Assessment Method (3)

- Evaluation
  - Evaluation step remains unchanged from MRP-146R0 and MRP-146S



DH Assessment Method (4)

- <u>Inspection</u>
  - For significant thermal cycling, inspection remains primary management strategy
  - Inspection frequency may be established with or without analysis
  - Without analysis, inspections every other refueling outage required



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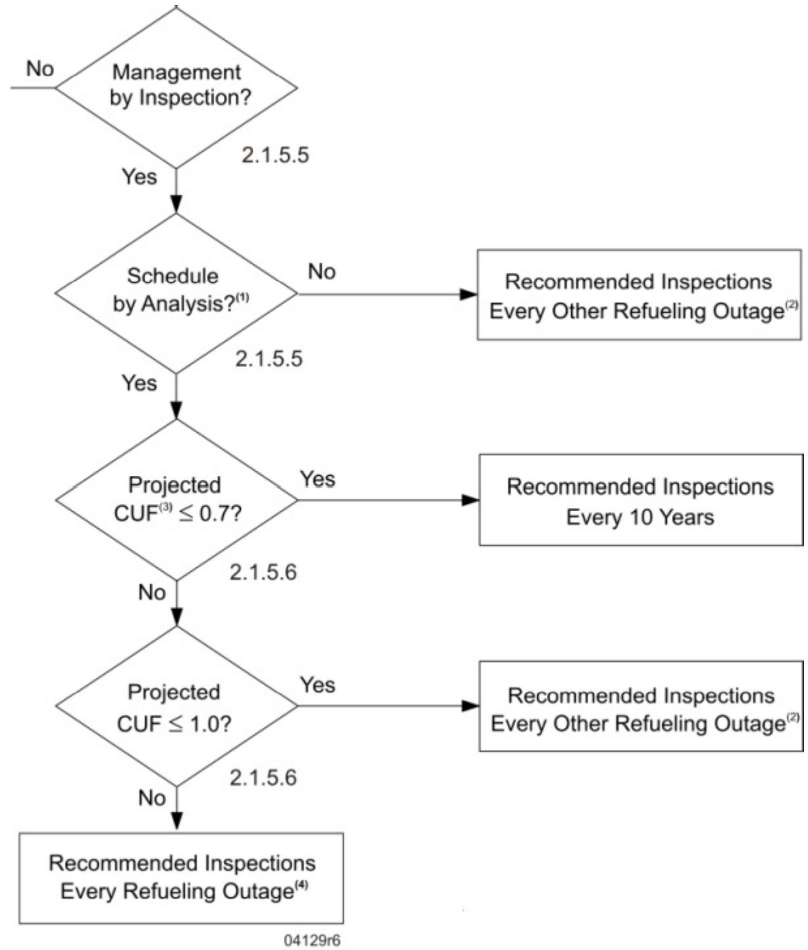
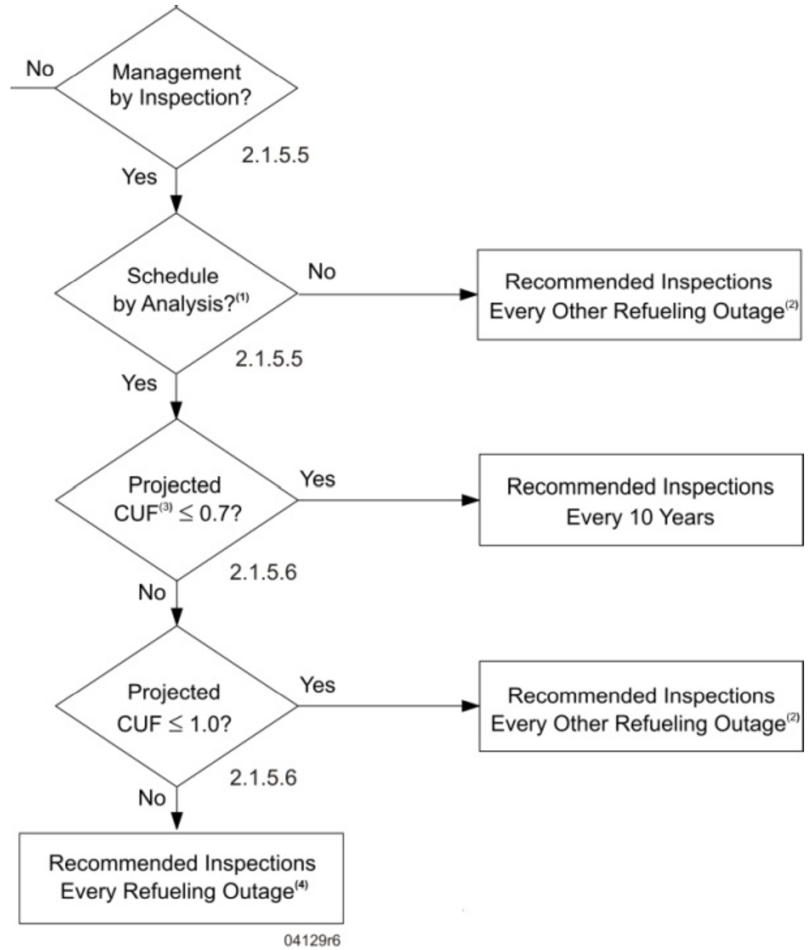
DH Assessment Method (5)

- Inspection (cont.)
  - Inspection frequency may be established using fatigue analysis or a Section XI Appendix L flaw tolerance evaluation (unchanged with MRP-146R1)
  - Projected CUF is defined as the CUF at the beginning of the RFO where inspection is planned



DH Assessment Method (6)

- Inspection (cont.)
  - Inspection frequency reduced somewhat with MRP-146R1 based on OE (details to follow)
  - For CUF > 1.0, more frequent inspection required
  - An item shall be entered into the plant's CAP indicating that the projected CUF > 1.0 for the affected location



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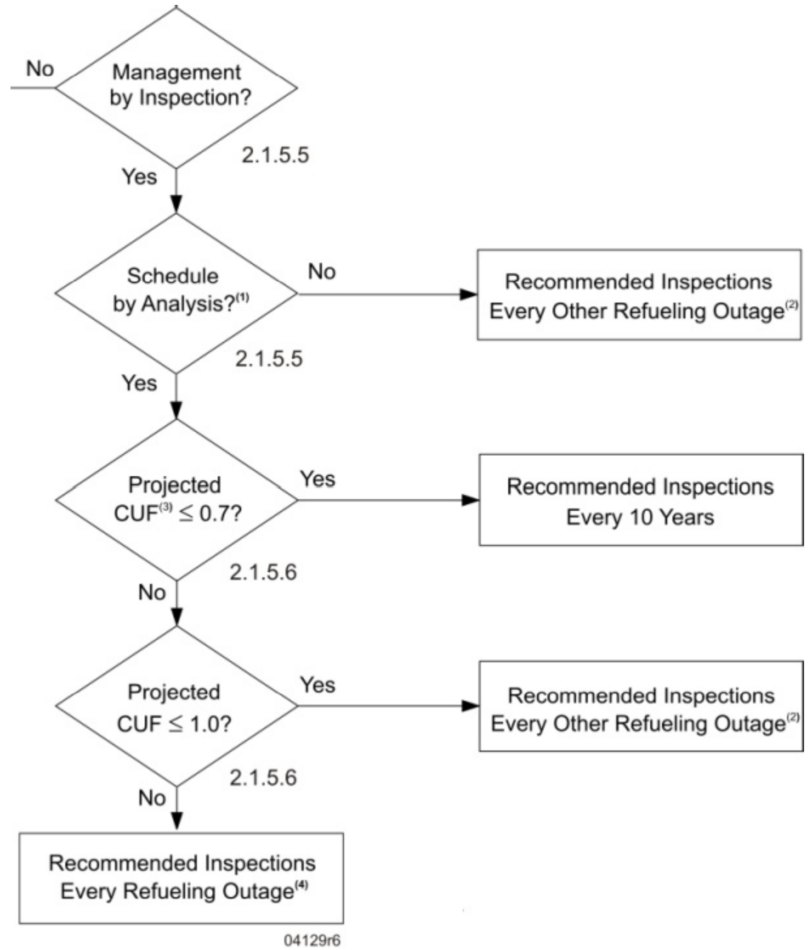
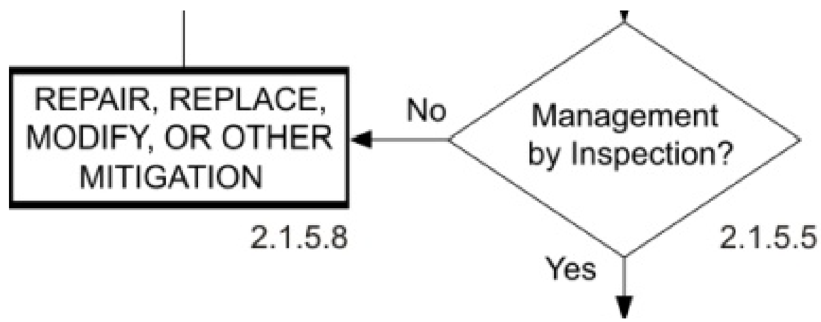
DH Assessment Method (7)

- Inspection (cont.)
  - Easing of inspection interval driven by:
    - Initial inspection findings
    - Time to failure slower than UH/H mechanism
    - Analysis methods are conservative (established in MRP-146S)



DH Assessment Method (8)

- Alternate Actions
  - Actions to mitigate thermal cycling loadings remains an option
  - Section 2.5 of MRP-146R1 significantly expanded to address thermal fatigue mitigation



Summary of Improved MRP-146R1 Guidance

MRP-146R1 Monitoring Guideline
General Monitoring Criteria

- Monitoring is generally undertaken for one of two reasons:
  - Verifying the absence of in-leakage
  - Demonstrating that the thermal loading is not as severe as predicted by analysis
- Since the amount of in-leakage could change with time, monitoring to detect in-leakage must be ongoing
- When monitoring for the presence of cycling in DH configurations, it is sufficient to take data during normal plant operation for one operating cycle (if the data is to be used to supplement analysis, monitoring may be removed after two operating cycles)

MRP-146R1 Monitoring Guideline
General Monitoring Criteria (2)

- Specific minimum duration and frequency requirements are provided in MRP-146R1
- Significant changes to RCS normal operating conditions (i.e., power up-rate) may require re-assessment of need for monitoring

MRP-146R1 Monitoring Guideline
Temperature Monitoring

- Temperature monitoring sensors are typically either:
  - Strap-on thermocouples
  - Resistance temperature detectors
- Surface contact with piping and being sufficiently insulated to avoid ambient effects important
- Obtaining accurate data has been an industry challenge – redundancy is highly recommended
- Many plants are still collecting monitoring data in response to NRC Bulletin 88-08 (guidance was limited)

MRP-146R1 Monitoring Guideline
Temperature Monitoring (2)

- Guidance given in MRP-146R1 is more prescriptive – data acquired for meeting 88-08 commitments may be used for MRP-146 actions provided MRP-146R1 requirements are met
- Temperature data taken on outside of pipe wall requires interpretation to determine fluid temperature
- Contributing factors include:
  - Frequency of fluid transient on inside of pipe
  - Thermal time lag through pipe thickness
  - Response attenuation by axial and circumferential heat transfer

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MRP-146R1 Monitoring Guideline

Valve Leakage Determination Guideline

- MRP-146R1 provides several methods that may be used to determine flow rates across leaking isolation valves in UH/H branches (e.g., safety injection or out-of-service charging lines)
- These methods involve either physical measurement of fluid flow or more sophisticated non-invasive technologies
- Alternate methods are acceptable
- As a point of reference, the leak tightness specification for isolation motorized gate valves is about 10 cc/hr (about two orders of magnitude less than the lower bound in-leakage rate of concern from MRP-132)

MRP-146R1 Inspection Guidelines

General Examination Requirements

- General examination requirements remain for the most part unchanged
- Requirement for examiners to be familiar with the unique aspects of inspection for thermal fatigue damage and for geometric considerations specific to small diameter piping now allows for alternate training methods beyond only the EPRI computer based training, MRP-36R1
- The EPRI NDE Center has thermal fatigue mock-ups available for utility training/practice

MRP-146R1 Inspection Guidelines
Inspection Volumes

- Several inspection volume changes and clarifications are made in MRP-146R1:
  - Base metal inspection requirement more clearly defined (for UH/H horizontal pipe sections)
  - Examination zone for socket-welded lines increased and more clearly defined
  - Two new figures added for clarity
- Examination guidance for elbow base metal and full penetration welds remains unchanged (for branches w/ butt-welded construction)

MRP-146R1 Thermal Fatigue Mitigation Guideline

- Thermal fatigue mitigation may be used to eliminate or reduce the potential or severity of future thermal fatigue cycling
- Significantly expanded guidance included in MRP-146R1
- Actions may include:
  - Plant modifications
  - Changes in plant operation
  - Preventative isolation valve maintenance
- Examples are provided for many of the actions described

MRP-146R1 Implementation and Next Steps

MRP-146R1 Implementation

- EPRI sponsored utility training for MRP-146R1 is underway (three sessions completed in 2011)
- Implementation of MRP-146R1 shall be complete by utilities and reflected in plant documentation as of the first RFO that initiates after January 31, 2012

Next Steps

- Currently, fracture mechanics analyses are being conducted in accordance with ASME Section XI, Appendix L to better understand piping flaw tolerance when subjected to swirl penetration cyclic stratification
  - Preliminary results show for a sample line where the fatigue usage is expected to be high, fatigue crack growth of a postulated flaw to an allowable depth would take ~ 4 years
  - An EPRI MRP report will be published detailing the methodology and providing examples – utility training will follow
- Heat transfer analysis regarding the interpretation of temperature monitoring data taken from the outside pipe surface is being considered

Conclusions

Conclusions

- MRP-146R1 allows for progressively more specific and rigorous evaluation as part of the assessment process
  - General screening
  - Determine significance of thermal fatigue potential
  - Inspection frequency based on severity of loading
- Many conservatisms inherent with each level
- MRP-146R1 provides utilities with the most current implementation guidance (replacing Rev. 0)

Conclusions (2)

- MRP-146R1 and supporting documents provide an effective approach to managing thermal fatigue in normally stagnant, non-isolable RCS branch lines
- PWR owners are using this approach moving forward
- EPRI committed to keeping the guidance current through future revision based on owner operating experience

Comments and Discussion

Attachment 8

REGULATORY RATIONALE BEHIND THE SRP 3.6.2/BTP 3-4 GUIDELINES FOR POSTULATING ASME CODE CLASS 1 PIPE BREAK LOCATIONS

One of the topics discussed during the January 5, 2012 public meeting on fatigue issues was related to the staff's guidelines included in Standard Review Plan (SRP) 3.6.2 and its associated Branch Technical Position (BTP) 3-4 guidelines for postulating ASME Code Class 1 pipe break locations. During the meeting, the NRC staff explained to the meeting participants the regulatory rationale behind these guidelines.

The regulatory basis for SRP 3.6.2 and its associated BTP 3-4 is 10 CFR Part 50, Appendix A, General Design Criterion 4 (GDC 4). GDC 4 requires that structures, systems, and components important to safety be designed to accommodate the effects of postulated accidents, including appropriate protection against the dynamic and environmental effects of a postulated pipe rupture.

SRP 3.6.2/BTP 3-4 provides, in part, guidelines acceptable to the staff for meeting these GDC 4 requirements. Specifically, it provides the screening criteria for plant designers to use for selecting pipe break locations. In applying the SRP guidelines, the postulated pipe break location is assumed to be a mechanistic failure, and the guidelines use stress and fatigue as deterministic criteria to identify postulated failure locations. A "mechanistic failure" is a postulated failure that is not initiated by any particular cause or event. The regulatory rationale behind those guidelines is based on ASME Section III design requirements and includes the postulation of a break location where the stress limit exceeds 80% of the applicable ASME Section III stress limit and where the fatigue limit exceeds a cumulative usage factor (CUF) of 0.1. The reduced stress limit provides a conservative margin for piping stresses due to causes or events that are not anticipated and often are not determinable such as those from unforeseen causes like operator errors or new degradation mechanisms.

With respect to the CUF, the SRP guidelines established 0.1 as its fatigue limit above which a break should be postulated. The larger margin adopted on CUF is to ensure an adequate margin on cyclic loadings and to take into account unanticipated conditions as well as uncertainties. In 1986, when the staff was revising its guidelines for postulating arbitrary intermediate breaks (which was later incorporated into GL 87-11, Relaxation in Arbitrary Intermediate Pipe Rupture Requirements), the staff also considered whether the CUF limit of 0.1 could be increased to 0.4 as recommended in ANSI/ANS 58.2 Standard . After some careful evaluation, the NRC staff determined that the CUF limit of 0.1 should not be increased to 0.4. The main reason was that several studies conducted for the NRC staff questioned the conservatism of the ASME design fatigue curve. Specifically, when the effects of environmental assisted fatigue (EAF) were not considered in the piping design, the calculation of the CUF might not be conservative. The staff also noted that when the effects of EAF are considered in the piping design, as has been proposed in several new-reactor standard plant designs, the staff would be willing to accept a CUF limit of 0.4 for postulating break locations due to fatigue.

The staff also noted that it has often revised its pipe break guidelines when a need exists. The staff discussed several cases when its pipe break guidelines were revised such as for facilitating inservice inspection of pipe welds, reducing excessive numbers of pipe whip restraints, and eliminating the operating basis earthquake loadings for new reactors. It was unclear to the staff whether there were any compelling reasons for the staff to revise its pipe break guidelines at this time since none were discussed. The staff also noted that the SRP provides guidelines (not requirements) acceptable to the staff for meeting its regulations. Other approaches might be acceptable provided they are adequately justified. The approach proposed by EPRI for selecting postulated fatigue pipe break locations based on using a risk-informed approach may be submitted by applicants or licensees wishing to use an alternative approach to that specified in the SRP. However, the staff noted that such a submittal would likely involve a lengthy review to ensure that a reasonable number of break locations are postulated to address uncertainties and unanticipated events.